WCAP-16996-P/WCAP-16996-NP Volumes I, II, and III, Revision 0, "Realistic Loss-Off-Coolant Accident [LOCA] Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM™ LOCA [FSLOCA] Methodology)"

REQUEST FOR ADDITIONAL INFORMATION (RAI)

EIGTH SET OF RAI QUESTIONS

RAI Questions 122 through 139

ENCLOSURE 5

The most current **Table 1** is contained in the proprietary Attachment to this package.

Table 2: List of Abbreviations

Abbreviation	Meaning	Note
1D	One Dimensional	
ADAMS	Agencywide Documents Access and Management System	
ASTRUM	Automated Statistical Treatment of Uncertainty Method	
BAF	Bottom of Active Fuel	
BNWL	Battelle Northwest Laboratories	
CCFL	Counter-Current Flow Limitation	
CFD	Computational Fluid Dynamics	
CHF	Critical Heat Flux	
COBRA	Coolant Boiling in Rod Arrays	
COCO	Containment Pressure Analysis Code	
COSI	Condensation on Safety Injection	
CQD	Code Qualification Document	
CSAU	Code Scaling, Applicability, and Uncertainty	
CSE	Containment Systems Experiment	
CT	Churn-Turbulent	
DC	Downcomer	
DEG	Double-Ended Guillotine	
EM	Evaluation Model	
ECCS	Emergency Core Cooling System	
EOP	Emergency Operating Procedures	
EPRI	Electric Power Research Institute	
FD	Film/Drop	
FLECHT	Full Length Emergency Cooling Heat Transfer	
FSLOCA	Full Spectrum Loss-of-Coolant Accident	
GE	General Electric	
HL	Hot Leg	
HPTF	High Pressure Test Facility	
HTSTR	Heat Structure	
IAEA	International Atomic Energy Agency	
ID	Inner Diameter	
ISP	International Standard Problem	
JAERI	Japan Atomic Energy Research Institute	
LB	Large Bubble	
IBLOCA	Intermediate Break Loss-of-Coolant Accident	
IET	Integral Effects Test	
LBLOCA	Large Break Loss-of-Coolant Accident	
LHGR	Linear Heat Generation Rate	
LOCA	Loss-of-Coolant Accident	
LSTF	Large Scale Test Facility	
MLO	Maximum Local Oxidation	
NRC	U. S. Nuclear Regulatory Commission	
OD	Outer Diameter	
ORNL	Oak Ridge National Laboratory	

Table 2: List of Abbreviations (Continued)

Abbreviation	Meaning	Note
PCT	Peak Cladding Temperature	
PDF	Probability Density Function	
PIRT	Phenomena Identification and Ranking Table	
PKL	Primärkreislauf (German for Primary Coolant Circuit)	
PWR	Pressurized Water Reactor	
PIRT	Phenomena Identification and Ranking Table	
RBHT	Rod Bundle Heat Transfer Test	
RCP	Reactor Coolant Pump	
RCS	Reactor Coolant System	
RELAP	Reactor Excursion Leak Analysis Program	
RG	Regulatory Guide	
ROSA	Rig-of-Safety Assessment	
RPV	Reactor Pressure Vessel	
SB	Small Bubble	
SBLOCA	Small Break Loss-of-Coolant Accident	
SEASET	Separate Effects and System Effects Test	
SG	Steam Generator	
SI	Safety Injection	
SLB	Small-to-Large Bubble	
TAF	Top of Active Fuel	
TC	Thermocouple	
TEE	T-Junction	
TF	Three-Field	
TF	Two-Fluid	
TF2	Three-Field and Two-Fluid	
THTF	Thermal Hydraulic Test Facility	
TPTF	Two-Phase Flow Test Facility	
TRAC	Transient Reactor Analysis Code	
TRAC-M	Transient Reactor Analysis Code - Modernized	
TRAM	Transient and Accident Managemen	
UPTF	Upper Plenum Test Facility	

RAI Question #122: Upper Bound Approach to Reactor Coolant Pump Trip Time during Loss-of-Coolant Accident with Offsite Power Available

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.3, "Reactor Accident Boundary Conditions," explains that "offsite power determines whether RCS pumps initially remain on, and whether pumped safety injection (and containment safeguards) come on with only valve opening and alignment delays. The effect of the RCS pumps on the LOCA transient may be significantly different, depending on whether they are assumed to coast down or continue running (until operator action is taken, if applicable)." Section 30, "Technical Basis of Statistical Procedures Applied in FSLOCA Uncertainty Methodology," Subsection 30.4, **[ ]** describes a statistical approach for treatment of Offsite Power Availability (OPA) at the time of a postulated small or large break LOCA event (OPA=ON or OPA=OFF) in the FSLOCA methodology.

In analyzing LOCAs with offsite power available, WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.5.2, "Variability of Plant Conditions Due to Operation Actions," summarizes that **[ ]** The upper bound to operator action time to trip the RCPs is taken as **[ ]** from the time reactor trip occurs, and this time is used in the reference break spectrum with offsite power available in Section 27, Volume 3 of this document." WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 27, "Reference Break Spectrum Analysis," describes representative LOCA analyses of small, intermediate, and large breaks for the reference V. C. Summer and Beaver Valley Unit 1 plants.

Please provide the following additional information regarding the approach to RCP trip time modeling with offsite power available as implemented in the FSLOCA methodology.

(1) The analysis provided in Subsection 25.5.2 considers the short-term phase of a SBLOCA and refers to EOPs specific to Indian Point Unit 2, a four-loop Westinghouse PWR. At the same time, Subsection 25.5.2 states that the upper bound to operator action time to trip the RCPs, assessed at **[ ]** from the time reactor trip occurs, is used in the LOCA analyses for large, intermediate, and small break sizes described in Section 27, "Reference Break Spectrum Analysis." Please clarify if the described approach based on an upper bound to operator action time to trip the RCPs set at **[ ]** from the time when reactor trip occurs is considered applicable on a generic basis for performing LOCA analyses with the FSLOCA methodology. Please provide a justification for the described approach.

(2) Please describe the rationale for introducing "an upper bound to operator action time to trip the RCPs" as part of the approach for simulating RCPs operation during LOCAs with offsite power available in the FSLOCA methodology. Explain how such "an upper bound to operator action time to trip the RCPs" can be credited for modeling RCPs operation with offsite power available in performing best-estimate analyses of both small and large break LOCA. Describe the technical basis that demonstrates the validity of the approach implemented in the FSLOCA methodology.

RAI Question #123: Factors Affecting Reactor Coolant Pump Trip Time for Loss-of-Coolant Accident Analyses with Offsite Power Available

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 2, "Evaluation Model Functional Requirements," Subsection 2.3.1, "LOCA Scenario Specification," recognizes the possibility of operator action in the accident scenario. Subsection 2.3.1 also explains that the availability of the RCPs following a reactor trip event is considered so that "variability in the pump trip time does exist."

Section 3.15, "Special Considerations for a Small-Break Loss-of-Coolant Accident in Pressurized Water Reactors," in the NRC RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, requires that "the pump operation assumptions used in the calculations should be the most likely, based on operating procedures, with appropriate consideration of the uncertainty of the pump operation during an actual event."

Please provide additional information regarding the approach to model RCP trip time with offsite power available as part of the FSLOCA methodology as follows.

(1) Explain specifically how the approach to model RCP trip time with offsite power available for LOCA analyses takes into consideration the following factors: (a) plant-specific design features, (b) relevant EOP requirements and applicable criteria, (c) break size, (d) plant monitored conditions, (e) availability, performance, and correct use of equipment, (f) operator recognition of the event and action, as well as any other factors considered of relevance.

(2) Please provide an assessment for the uncertainties associated with each individual factor identified in Item (1) above with regard to the RCP trip time. Please consider each factor individually and assess its contribution to the overall uncertainty in the RCP trip time. Include appropriate ranges and associated uncertainties. Present examples of assessment results for PWR plant types with similar design features, EOP requirements, and other relevant factors, for which the FSLOCA methodology is considered applicable. Include a table that documents the assessed time contribution to RCP trip time due to each individual factor, the RCP trip time uncertainty range, and the RCP trip time credited in LOCA analyses. If appropriate, please provide a separate table for each plant type. Describe the introduced assumptions, explain and discuss the analysis results, and compare the assessments.

RAI Question #124: Reactor Coolant Pump Trip Time for Small Break Loss-of-Coolant Accident Analyses with Offsite Power Available

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.5.1, "EOP Sequences for a Small Break LOCA," summarizes EOPs relevant to the short-term phase of a small break LOCA for Indian Point Unit 2. The procedures are summarized in Table 25-1, "Condensed EOPs for Indian Point Unit 2, Short-Term Portion." In particular, it is explained that **[**

] The RCP trip time variability is caused by the time the operators take to identify that the trip conditions exist during their periodic scan of system parameters, and the small increment required to actually perform the trip." Subsection 25.5.1 also states that "plants typically follow a generic template for the generation of EOPs, and plant-to-plant differences in the EOP structure are not expected to be important for the purpose of performing a LOCA safety analysis."

Please provide the following additional information regarding the approach to RCP trip time modeling with offsite power available as implemented in the FSLOCA methodology for performing small break LOCA analyses.

(1) Explain how the approach to modeling RCPs trip due to operator action in determining the boundary conditions for SBLOCA analyses is applied on a plant-by-plant basis in the FSLOCA methodology. In particular, identify relevant contributing factors and types of plant-specific information (for example, available data, EOPs and criteria, etc.) that are considered for modeling RCP trip time on a plant-by-plant basis in individual plant analyses. Provide and explain the uncertainty and significance, in terms of having an impact on the credited RCPs trip time, associated with such factors and plant-specific information.

(2) Provide examples of assessed RCPs trip times considered applicable for small break LOCA analyses and determined for different PWR plant types based on design features, EOP requirements, and other pertinent factors. Present examples of assessment results for PWR plant types for which the FSLOCA methodology is considered applicable. Include a table that documents the assessment results for the examined cases. Describe the introduced assumptions, explain and discuss the analysis results, and compare the assessments.

(3) The upper bound to operator action time to trip the RCPs of **[ ]** from the time when reactor trip occurs, is provided in Subsection 25.5.2, "Variability of Plant Conditions Due to Operation Actions," on the basis of small break LOCA considerations. Please provide the plant-specific values for the RCPs trip times used in the demonstration plant SBLOCA analyses for the FSLOCA methodology presented in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping And Sensitivity Studies," and in Section 31, "Full Spectrum™ LOCA Demonstration Analysis." Describe the introduced assumptions, explain and discuss the applicability of the analysis results, and compare the assessments. Explain the basis for the used RCPs trip times considering the response to the item (1) above.

RAI Question #125: Reactor Coolant Pump Trip Time for Large Break Loss-of-Coolant Accident Analyses with Offsite Power Available

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping and Sensitivity Studies," Subsection 28.1.2, "Offsite Power Availability – LBLOCA," analyzes effect of offsite power availability on RCPs behavior for LBLOCA analyses performed with WCOBRA/TRAC-TF2 for the reference V. C. Summer (CGE) and Beaver Valley Unit 1 (DLW) plants. Subsection 28.1.2 explains that "with LOOP, the RCP trip is modeled coincident with the reactor trip at the beginning of the transient, so the pumps in the intact loops coast down while the pump in the broken loop is accelerated by the flow toward the break." In the case "with offsite power available (OPA), the pumps continue to rotate at a fixed speed until operator trip."

The RCP rotational speeds in the intact loops and in the broken one the V. C. Summer (CGE) nominal DEG break demonstration plant analysis are shown in Figure 28.1.2-2, "Intact Loop Pump Speed, CGE Offsite Power Availability Sensitivity," and in Figure 28.1.2-3, "Broken Loop Pump Speed, CGE Offsite Power Availability Sensitivity," respectively. For the Beaver Valley Unit 1 (DLW) nominal DEG break demonstration plant analysis, the RCP rotational speeds in the intact loops and in the broken one are shown in Figure 28.1.2-8, "Intact Loop Pump Speed, DLW Offsite Power Availability Study," and in Figure 28.1.2-9, "Broken Loop Pump Speed, DLW Offsite Power Availability Study," accordingly. As it can be scaled from these graphs, the RCPs pump speeds in all loop start decreasing at about **[ ]** following the break initiation in the analyses with offsite power available for both plants.

Please provide the following additional information regarding the approach to RCP trip time modeling with offsite power available as implemented in the FSLOCA methodology for performing large break LOCA analyses.

(1) Explain how the approach to modeling RCPs trip due to operator action in determining the boundary conditions for large break LOCA analyses is applied on a plant-by-plant basis in the FSLOCA methodology. In particular, identify relevant contributing factors and types of plant-specific information (for example, available data, EOPs and criteria, etc.) that are considered for modeling RCP trip time on a plant-by-plant basis in individual plant analyses. Provide and explain the uncertainty and significance, in terms of having an impact on the credited RCPs trip time, associated with such factors and plant-specific information.

(2) Provide examples of assessed RCPs trip times considered applicable for large break LOCA analyses and determined for different PWR plant designs based on design features, EOP requirements, and other pertinent factors. Present examples of assessment results for PWR plant types for which the FSLOCA methodology is considered applicable. Include a table that documents the assessment results for the examined cases. Describe the introduced assumptions, explain and discuss the applicability of the analysis results, and compare the assessments.

(3) Please provide the plant-specific values for the RCPs trip times used in the demonstration plant large break LOCA analyses for the FSLOCA methodology presented in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping And Sensitivity Studies," and in Section 31, "Full Spectrum

LOCA Demonstration Analysis." Describe the introduced assumptions, explain and discuss the analysis results, and compare the assessments.

(4) The upper bound to operator action time to trip the RCPs of **[ ]** from the time when reactor trip occurs, provided in Subsection 25.5.2, "Variability of Plant Conditions Due to Operation Actions," on the basis of small break LOCA considerations, agrees closely with the timing of about **[ ]** following the break initiation when the RCP pump speeds were predicted to start decreasing with offsite power available in the large break demonstration plant analyses presented in Subsection 28.1.2. Please explain the basis for the implemented RCPs trip time considering the response to Item (1) above.

RAI Question #126: Delay in Operator Action to Trip Reactor Coolant Pumps during Small Break Loss-of-Coolant Accidents

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.5.2, "Variability of Plant Conditions Due to Operation Actions," explains that **[**

]

(1) Please clarify if the above identified range of **[ ]** expected in operator action to trip the RCP for small break LOCAs is considered applicable on a generic basis in performing LOCA analyses using the FSLOCA methodology.

(2) Regarding the operator action to trip the RCPs during SBLOCAs, Subsection 25.5.2 clarifies that **[**

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Please identify the "studies" described as **[ ]** in the above provided citation form Subsection 25.5.2 and provide references for the source documents containing these studies. In particular, provide details and explain how the identified **[ ]** demonstrates the acceptability and appropriateness of the provided delay time of **[ ]** needed for operator action to trip the RCPs during a LOCA event.

(3) Please explain if uncertainty in human reliability was considered in the approach to model the operator action to trip the RCPs implemented in the FSLOCA methodology.

(4) If the approach for determining the delay time needed for operator action to trip the RCPs during a LOCA event will be applied in FSLOCA methodology on a plant-by-plant basis, please identify and describe the type of plant-specific information that will be considered in individual plant applications.

RAI Question #127: Single Failure Assumptions in Loss-of-Coolant Accident Analyses

Criterion 35, "Emergency Core Cooling," in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 to Code of Federal Regulation Part 50, requires that a single failure be assumed when analyzing safety system performance. The NRC RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, states that "Appendix A to 10 CFR Part 50 requires that a single failure be considered when analyzing safety system performance and that the analysis consider the effect of using only onsite power and only offsite power."

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.3, "Reactor Accident Boundary Conditions," in its part "Single Failure Assumption," states that "the loss of a train may be assumed for the determination of pumped ECCS flow during the LOCA, while the train will be assumed to operate in the calculation of containment backpressure." It is also explained that "alternatively, a more complete analysis using consistent assumptions may be performed on a plant-by-plant basis."

(1) Please provide a table that lists single failure assumptions considered possible and applicable for the purposes of LOCA analyses using the FSLOCA methodology. Please use a separate raw for each combination and formulate the single failure assumptions in two separate columns. Include the single failure assumption applicable for modeling the RCS response with WCOBRA/TRAC-TF2 in the first column and the one applicable for the containment backpressure modeling in the second column. Please consider small and large break LOCA analysis applications. Explain the identified single failure assumptions along with pertinent conditions and supporting considerations.

(2) Subsection 25.3, "Reactor Accident Boundary Conditions," in its part "Single Failure Assumption," states that "a more complete analysis using consistent assumptions may be performed on a plant-by-plant basis." Please explain what such "a more complete analysis," when "performed on a plant-by-plant basis," includes. Identify and describe the "consistent assumptions" that are considered applicable for performing the analysis. Describe the types of plant-specific information that are considered in individual plant analyses. If failure-related assumptions that can be identified and used in analyses on a on a plant-by-plant basis are not included in the response to Item (1) above, please provide a table that describes them. Explain the identified single failure assumptions along with pertinent conditions and supporting considerations.

RAI Question #128: Prolonged RCP Operation during Small Break Loss-of Coolant Accidents

J. Gonzalez-Cadelo, et al., "Applying Integrated Safety Assessment Methodology, Analysis of Cold Leg SBLOCA With Failed HPSI," 21st International Conference Nuclear Energy for New Europe Ljubljana 2012, September 5-7, Ljubljana, Slovenia, provided results from the SBLOCA study of predicted PWR PCTs as a function of the break size and the RCPs trip delay. The work was performed for the Almaraz Unit I Nuclear Power Plant (NPP), a Westinghouse three-loop PWR, using the U.S. NRC code TRACE. Small breaks in the cold leg ranging between 1 inch and 5 inches of equivalent break diameter were analyzed with an increment as low as 0.25 inch. RCP trip delays ranging between zero and up to 10,000 seconds (~167 minutes) were examined with a time interval as low as 500 seconds. The study referenced Westinghouse EOPs EOP E-0, EOP E-1, and EOP ES-1.2 along with pertinent foldout pages related to small break LOCA sequences with early secondary-side depressurization and uncertain RCP trip time due to assumed failure of High Pressure Safety Injection (HPSI). The reported results exhibited a damage domain trend with PCTs in excess of 2,200 ºF (1,204.4 ºC or 1,477.6 K) at relatively short RCP trip times (about 500 seconds or less), for cold leg breaks between 2.5 inches and 3 inches in diameter.

(1) Please identify and describe SBLOCA sequences and accident conditions under which RCPs can remain in operation for prolonged time periods. Such sequences can be characterized with large uncertainties in RCP trip time. In identifying and describing these LOCA sequences and related accident conditions, please provide consideration of initial plant conditions, plant design and safety features, availability and performance of equipment, single failure assumptions, EOP requirements, and other relevant factors.

(2) Please explain how LOCA sequences, as identified in the response to Item (1) above, are accounted for in analyzing SBLOCAs using the FSLOCA methodology.

(3) Please describe the analyses that established the technical basis for prolonged RCP operation for the LOCA transients identified in the response to Item (1) above. Explain how corresponding EOP requirements and criteria related to RCP operation and trip were developed. Summarize and present the results from such analyses and identify those performed with WCOBRA/TRAC-TF2, as applicable.

(4) Please provide WCOBRA/TRAC-TF2 prediction results for a spectrum of break sizes that analyze SBLOCAs with prolonged RCP operation. In addition, please compare these predictions against results from WCOBRA/TRAC-TF2 analyses in which RCPs were tripped to illustrate the impact of RCP operation.

RAI Question #129: Early RCP Trip during Small Break Loss-of-Coolant Accidents

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.5.1, "EOP Sequences for a Small Break LOCA," describes continuously monitored conditions that are used in EOP E-0 "Reactor Trip or Safety Injection" for comparison against established RCP trip criteria when determining boundary conditions assumed for small break LOCA analysis calculations. It is mentioned in Subsection 25.5.1 that although the description is specific to Indian Point Unit 2, "plants typically follow a generic template for the generation of EOPs."

(1) Please identify and describe SBLOCA sequences and accident conditions under which RCPs are tripped early in the transient. In identifying and describing these LOCA sequences and related accident conditions, please provide consideration of initial plant conditions, plant design and safety features, availability and performance of equipment, single failure assumptions, EOP requirements, and other relevant factors.

(2) Please describe the analyses that established the technical basis for tripping RCPs in the class of LOCA transients identified in the response to Item (1) above. Explain how corresponding EOP requirements and criteria related to RCP operation were established in the generic template used for generation of EOPs. Summarize and present the results from such analyses and identify those performed with WCOBRA/TRAC-TF2.

(3) Please provide WCOBRA/TRAC-TF2 prediction results for a spectrum of break sizes that analyze SBLOCAs with RCPs tripped early in the transients. In addition, please compare these predictions against results from WCOBRA/TRAC-TF2 analyses in which RCPs were not tripped to illustrate the impact of RCP operation.

RAI Question #130: Break Location Impact in Previous Small Break Loss-of-Coolant Accident Analyses

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 2, "Evaluation Model Functional Requirements," Subsection 2.3.1, "LOCA Scenario Specification," states: "Sensitivity studies based on Appendix K methods have identified a cold leg break to be the most limiting in terms of location. Since this was also true of earlier best-estimate cases analyzed, this is taken to be the limiting break location for the following PIRT discussion."

(1) Please identify and describe the conservative analyses of the effect of break location described as "sensitivity studies based on Appendix K methods" in the above given citation from Subsection 2.3.1. Summarize the findings from the performed analyses and present major prediction results using tables and graphs as appropriate. Include description of relevant modeling assumptions and computer codes used. Also, please provide a list of references identifying the source documents containing these studies.

(2) Please identify and describe the best-estimate studies analyzing the effect of break location and described as “earlier best-estimate cases analyzed” in the above given citation from Subsection 2.3.1. As for Item (1) above, please summarize the findings from the performed analyses and present major prediction results using tables and graphs as appropriate. Include description of relevant modeling assumptions and computer codes used. Provide a list of references identifying the source documents containing these analyses.

RAI Question #131: Break Location Impact in Small Break Loss-of-Coolant Accident Analyses Using WCOBRA/TRAC-TF2

As pointed out in Section 4.4.2, “Break Location,” in NUREG-0623 (see B. Sheron, “Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors,” NUREG-0623, November 1979), vendor analyses identifying limiting breaks with regard to break location led to different findings. Thus, Westinghouse “concluded that a break in the cold leg discharge piping with the pumps running resulted in the limiting consequences” whereas Combustion Engineering concluded that “the breaks postulated in the hot leg with delayed pump trip or pumps running were the most limiting with regard to peak cladding temperatures.” Also, no best-estimate analyses of small break LOCAs by Westinghouse were identified in Section 4.4.6, “Best-Estimate Analysis,” in NUREG-0623.

Section 3.15, “Special Considerations for a Small-Break Loss-of-Coolant Accident in Pressurized Water Reactors,” in U.S. NRC RG 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” May 1989, recognizes that “break flow may be greatly influenced by the location and specific geometry of the break.” Accordingly, “small-break loss-of-coolant accident calculations should, therefore, include various assumed break locations in the spectrum of breaks analyzed.”

Please provide the following additional information regarding WCOBRA/TRAC-TF2 analyses performed to examine sensitivity of code predictions to break location as part of substantiating and confirming the approach for treatment of break location for small break LOCA analyses implemented in the FSLOCA methodology.

(1) Please identify best-estimate studies performed with WCOBRA/TRAC-TF2 to analyze the effect of break location in small break LOCAs. Provide a list of references that identify the source documents containing these analyses.

(2) Please provide and describe WCOBRA/TRAC-TF2 prediction results for a spectrum of break sizes that analyze small breaks occurring in a cold leg piping between the RCP and the reactor pressure vessel as well as in a hot leg. For breaks located in a hot leg, please consider possible sensitivity with regard to the location of the pressurizer vessel connection. Include description of relevant modeling assumptions and the computer code versions used.

(3) Present direct comparisons of prediction results for key parameters from small break LOCA analyses simulating transients with the same break sizes and different break locations. Present such comparison plots for different PWR plant types that will be analyzed with the FSLOCA methodology.

(4) Provide and describe WCOBRA/TRAC-TF2 prediction results that examine the impact of RCPs trip delay with offsite power available on the limiting break location for small break LOCAs. Include analyses of small break LOCA sequences with RCPs tripped early in the transient and such with prolonged RCPs operation.

RAI Question #132: Steam Generator Decay Heat Removal during Small Break Loss-of-Coolant Accidents

The main steam safety valves (MSSVs) are direct-acting valves (actuated only by pressure) that provide overpressure design protection and backup decay heat removal capability when the steam dumps and secondary atmospheric dump valves cannot be used. Each main steam line has several safety valves with staggered set pressures to provide an increased relieving capacity with an increasing overpressure. As described by T. E. Wierman, et al., "Industry Performance of Relief Valves at U.S. Commercial Nuclear Power Plants through 2007," NUREG/CR-7037, March 2011, the set pressures for these valves are a nominal 1170, 1200, 1210, 1220, and 1230 psig with the highest setpoint being less than 110 percent of the SG design pressure.

WCAP-16996-P/WCAP-16996-NP, Volumes I, II and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.3, "Reactor Accident Boundary Conditions," in summarizing the modeling approach with regard to the SG secondary side boundary conditions, states that **[**

] Specifically, it is explained that **[**

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Please provide additional information related to the modeling of SG secondary conditions in WCOBRA/TRAC-TF2 plant analyses to predict decay heat removal via primary to secondary heat transfer in the SGs during SBLOCAs as follows.

(1) State the method of decay heat removal due to heat transfer from the RCS to the SG secondary sides in SBLOCA analyses with WCOBRA/TRAC-TF2 as credited in the FSLOCA methodology. Identify the credited systems, conditions for their operation, and introduced assumptions as applied in modeling decay heat removal via the SGs in SBLOCA analyses.

(2) Describe the approach to determine the set pressures for the MSSVs. In particular, please explain how the implemented approach takes into consideration and models the following factors: (a) uncertainty in the setpoint characteristics of the safety valves, (b) pressure drop from the SG to the safety valves, (c) uncertainty in rated relief capacities of the safety valves, and (d) criteria for using the next highest pressure setpoint. For example, Section 7.2.2, "Westinghouse-Designed Plants," in NUREG-0623 (see B. Sheron, "Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," NUREG-0623, November 1979), in discussing the system pressure for manual pump trip, describes that the next highest pressure setpoint for the secondary system safety valves is used when the calculated relief flow is greater than 60 percent of the rated valve capacity at the previous pressure setpoint.

(3) Present a table that documents the parameters identified in Item (2) above and used in the demonstration plant small break LOCA analyses for the FSLOCA methodology presented in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping And Sensitivity Studies," and in Section 31, "Full Spectrum LOCA Demonstration Analysis." Subsection 31.1.1, "Break Area Ranges," identifies only a quantity described as "the lowest MSSV set pressure" and provides its value as **[ ]** for the V. C. Summer case analysis.

RAI Question #133: Steam Generator Heat Transfer Modeling

According to Section 3.2.7, "Primary to Secondary Heat Transfer (Not Applicable to Boiling Water Reactors)," in the NRC RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989, "heat transferred between the primary and secondary systems through the steam generators should be considered in the calculation and should be calculated in a best-estimate manner."

(1) Please describe the mechanisms that participate in the primary to secondary heat transfer through the SG heat exchange tubes. Identify factors that can have an impact on the primary to the secondary heat transfer mechanisms during normal plant operation and under LOCA conditions. Include consideration of deposits fouling of SG heat transfer tubes and supporting structures and associated effects on the overall heat resistance and thermal performance degradation.

(2) Please present a table that identifies the correlations used in WCOBRA/TRAC-TF2 to model the heat transfer mechanisms on the outer side of the SG heat transfer tubes. Identify the experimental database for each provided correlation and provide its applicability range. Compare the applicability ranges for the correlations against typical conditions expected during LOCAs. Explain how the factors affecting the heat transfer mechanisms as identified in Item (1) above are accounted for in the implemented heat transfer models. Summarize the technical basis for these models along with the supporting data and analyses. In particular, consider the effect of thermal performance degradation due to deposits fouling.

(3) Please present a table that identifies the correlations used in WCOBRA/TRAC-TF2 to model the heat transfer mechanisms on the inner side of the SG heat transfer tubes. Identify the experimental database for each provided correlation and provide its applicability range. Compare the applicability ranges for the correlations against typical conditions expected during LOCAs. Explain how the factors affecting the heat transfer mechanisms as identified in Item (1) above are accounted for in the implemented heat transfer models. Summarize the technical basis for these models along with the supporting data and analyses. In particular, consider the effect of non-condensable gas on heat transfer inside the SG tubes.

RAI Question #134: Steam Generator Tube Plugging Levels

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.1, "Plant Physical Configuration," in summarizing the modeling approach with regard to SG tube plugging, states that **[**

]

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping and Sensitivity Studies," Subsection 28.1.6, "Steam Generator Hydraulics: Tube Plugging – LBLOCA," present results for tube plugging levels of 0 percent, 10 percent, and 20 percent for a nominal double-ended guillotine break demonstration plant analyses for V. C. Summer, a three-loop Westinghouse PWR plant.

Please provide the following additional information regarding the approach to SG tube plugging modeling implemented in the FSLOCA methodology for performing LOCA analyses.

(1) Please explain how the **[ ]** SG tube plugging fraction is established and used in determining relevant boundary conditions for performing LOCA analyses using the FSLOCA methodology.

(2) WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping and Sensitivity Studies," referring to **[**

] discussed in Section 28.1.6 and in Section 28.2.9 respectively, states that **[ ]** Please identify and describe the type of plant-specific information that is considered in determining "a plant-specific **[ ]** for the SG tube plugging used in LOCA and associated uncertainty analyses. In particular, please explain SG design-specific information and SG operational history are taken into account when determining the **[ ]** SG tube plugging fraction on a plant-specific basis.

(3) Please provide the information identified in Item (2) above and the plant-specific **[ ]** for the SG tube plugging levels used in the demonstration plant analyses for the FSLOCA methodology presented in WCAP-16996-P/ WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping And Sensitivity Studies," and in Section 31, "FSLOCA Demonstration Analysis." Describe the introduced assumptions, explain and discuss the applicability of the analysis results, and compare the assessments.

RAI Question #135: Steam Generator Tube Plugging Impact on Core Flow Stagnation for Large Break Loss-of-Coolant Accidents

Considering the modeling approach with regard to SG tube plugging, WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 29, "Assessment of Uncertainty Elements," Subsection 29.3.1, "Bounded Parameters," explains that **[**

]

Recognizing an additional phenomenon of importance for LBLOCA analyses, Subsection 29.3.1, "Bounded Parameters," acknowledges that **[**

] Specifically, Subsection 29.3.1 explains that **[**

]

(1) Please identify and describe the deterministic studies of SG tube plugging level that are referred to in the above provided citation from Subsection 29.3.1. Summarize and present results from such analyses that examine the impact of the assumed SG tube plugging level on large break LOCA predictions with a focus on core flow stagnation and possible flow reversal. Include plots that capture such predicted conditions for the core and the hot channel in particular. Analyze and show identified effects of flow stagnation on prediction results taking into consideration applicable safety criteria.

(2) Please identify best-estimate studies that have been performed with WCOBRA/TRAC-TF2 to analyze the effect of SG tube plugging on LBLOCA predictions associated with core flow stagnation and possible flow reversal in the core. Provide a list of references that identify the source documents containing these analyses. In addition to the analyses discussed in Subsection 28.1.6, "Steam Generator Hydraulics: Tube Plugging – LBLOCA," please present results from calculations obtained with WCOBRA/TRAC-TF2 that examine the impact of SG tube plugging on large break LOCA predictions due to core flow stagnation and possible flow reversal in the core.

(3) In presenting the analyses requested in Item (2) above, please provide additional information addressing the following items: (a) the analyses should cover a spectrum of large break sizes over which the examined effect takes place and code predictions show most sensitivity to break change, if so observed, and (b) the analyses should cover the entire range of possible SG tube plugging levels. Please provide consideration of different PWR plant types and show predictions results for a PWR plant design for which the sensitivity to SG tube plugging due to core flow stagnation is expected to be most pronounced, as applicable. Show the impact on prediction results with relevance to the safety criteria and include zoomed plots of computed results over a time window during which the stagnation effect is predicted to occur, as applicable.

RAI Question #136: Reactor Coolant Pump Trip Impact on Core Initial Thermal-Hydraulic Response for Large Break Loss-of-Coolant Accidents

The time of RCP trip following a LBLOCA can have an impact on the initial thermal-hydraulic response in the reactor pressure vessel and in the reactor core region in particular. Among others, processes such as flow stagnation and possible flow reversal in the core can be affected by RCPs operation and time of RCP trip. WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping and Sensitivity Studies," Subsection 28.1.2, "Offsite Power Availability – LBLOCA," presents LBLOCA analyses performed with WCOBRA/TRAC-TF2 for the reference V. C. Summer (CGE) and Beaver Valley Unit 1 (DLW) plants. The analyses examine the impact of offsite power availability on large break LOCA results.

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.5.2, "Variability of Plant Conditions Due to Operation Actions," states that **[**

]

Please provide the following additional information regarding WCOBRA/TRAC-TF2 analyses performed to examine sensitivity of the initial reactor core thermal-hydraulic response to RCP trip time with offsite power available.

(1) Please identify best-estimate studies that have been performed with WCOBRA/TRAC-TF2 to analyze sensitivity of large break LOCA predictions to RCP trip time with offsite power available with a focus on the initial thermal-hydraulic response in the reactor pressure vessel and in the reactor core region. Provide a list of references that identify the source documents containing these analyses. In addition to the analyses discussed in Subsection 28.1.2, "Offsite Power Availability – LBLOCA," please present results from WCOBRA/TRAC-TF2 calculations to examine sensitivity to RCP trip time following a large break LOCA.

(2) Please provide WCOBRA/TRAC-TF2 prediction results analyzing sensitivity of LBLOCA predictions to RCP trip time with offsite power available with a focus on the effect of core flow stagnation and possible flow reversal in the core.

(3) In presenting the analyses requested in Item (1) above, please provide additional information addressing the following items: (a) the analyses should cover a spectrum of large break sizes over which the examined effect takes place and code predictions show most sensitivity to break change, if so observed, and (b) the analyses should cover a range of RCPs trip times that reflects the uncertainty associated with the pump trip event. Please provide consideration of different PWR plant types and show predictions results for a PWR plant design for which the sensitivity of examined parameters to RCPs trip time is most pronounced, as applicable. Show the impact on prediction results with relevance to the applicable safety criteria and include zoomed plots of computed results over a time window during which the stagnation effect is predicted to take place, as applicable.

RAI Question #137: Steam Generator Tube Plugging Impact on Steam Generator Reversed Heat Transfer for Large Break Loss-of-Coolant Accidents

Considering the modeling approach with regard to SG tube plugging, WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 29, "Assessment of Uncertainty Elements," Subsection 29.3.1, "Bounded Parameters," explains that **[**

]

Following a large break LOCA, liquid entrainment from the upper plenum through the hot legs and into the SG tube bundles can lead to significant evaporation of entrained liquid due to reversed heat transfer from the hot SG secondary systems to the primary side of the SG tubes. In turn, this process can cause a primary pressure increase thus impacting the core thermal response. Such reversed heat transfer through the SGs can be dependent on the available SG heat transfer area, which is directly affected by the assumed SG tube plugging level.

(1) Please present results from calculations obtained with WCOBRA/TRAC-TF2 that examine the effect of the assumed SG tube plugging level on LBLOCA predictions with a focus on the effect of reversed heat transfer from the hot SG secondary systems and evaporation of entrained liquid on the SG tubes primary side.

(2) In presenting the analyses requested in Item (1) above, please provide additional information addressing the following items: (a) the analyses should cover a spectrum of large break sizes over which the examined effect takes place and code predictions show most sensitivity to break change and (b) the analyses should cover the entire range of possible SG tube plugging levels. Please provide consideration of different PWR plant designs and show predictions results for a PWR plant design for which the sensitivity of primary pressure increase to SG tube plugging is expected to be most pronounced, as applicable. Show the impact on prediction results with relevance to the applicable safety criteria and include zoomed plots of computed results over a time window during which the examined effect is predicted to take place, as applicable.

RAI Question #138: Safety Injection Pump Flow during Loss-of-Coolant Accidents

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 25, "Plant Sources of Uncertainty," Subsection 25.3, "Reactor Accident Boundary Conditions," in summarizing the modeling approach with regard to the SI flow, states that **[**

] Specifically, it is stated that "safety injection (SI) flow varies depending on the single failure assumed, and on the specific plant pump and injection line configuration. Current methods, which are also used in currently accepted evaluation models, provide conservative estimates of minimum and maximum flow, which take into account several uncertainties."

Please provide additional information related to the modeling of pumped SI flow in performing LOCA analyses using WCOBRA/TRAC-TF2 as identified below.

(1) Describe the approach to determine the **[ ]** that are assumed for LOCA analyses on a plant-specific basis. In particular, please identify and describe these "several uncertainties" that are taken into consideration by the "current methods, which are also used in currently accepted evaluation models" in order to provide conservative estimates as stated in Subsection 25.3, "Reactor Accident Boundary Conditions." In addition, please explain how the implemented approach accounts for and models the following factors: (a) uncertainties in safety injection pump characteristics, (b) injection line configuration, and (c) flow resistance and pressure drop along the injection lines.

(2) Please document the parameters identified in Item (1) above and used in the demonstration plant analyses of the FSLOCA methodology presented in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 28, "Scoping and Sensitivity Studies," and in Section 31, "FSLOCA Demonstration Analysis." State the assumptions and relevant conditions, as implemented in assessing parameters related to the SI pump flow modeling in the performed LOCA analyses. Include graphs of all pump flow characteristics and tables documenting assessments for other relevant parameters, as appropriate.

RAI Question #139: Asymmetrical Predictions in Modeling of Parallel Flow Configurations

Thermal hydraulic system codes, such as RELAP5, have been found to predict asymmetrical results when modeling parallel flow configurations, usually under low-flow conditions. Recognition of such modeling difficulties is presented by G. W. Johnsen, "RELAP5-3D Development & Application Status," Presentation at the 2002 RELAP5 International User's Seminar, September 4-6, 2001, Park City, Utah. In PWR plant analysis, such flow configurations can be related to parallel flow paths representing the cold legs in the same primary coolant loop of a Combustion Engineering (CE) PWR plant, parallel flow channels representing different azimuthal sections of a reactor vessel downcomer, such representing steam generator secondary side volumes or other regions of the reactor system. A possible solution approach in modeling a simple flow problem between parallel pipes is discussed by D. Lucas, "Recirculating Flow Anomaly Problem Solution Method," Proceedings of 8th International Conference on Nuclear Engineering ICONE8, Paper ID 8479, April 2-6, 2000, Baltimore, Maryland.

Please show that WCOBRA/TRAC-TF2 does not predict anomalous behaviors as described above for other codes when using three- and one-dimensional components. As part of the response, present predictions for an illustrative parallel pipe flow problem as implemented in the RELAP5 dual pipe flow input model presented below.

```
=Flow Anomaly Test Problem
*
*-----------------------------------------------------------------
*crdno          problem type       problem option
0000100              new              transnt
*-----------------------------------------------------------------
*crdno          input units          output units
```

```
0000102             british                 british
*-----------------------------------------------------------------------
*crdno    time 1     time 2
0000105   10.        40.     10000.
*-----------------------------------------------------------------------
0000110  nitrogen
*-----------------------------------------------------------------------
*crdno end time  min dt   max dt  control  minor ed  major ed  restart
0000201   5000.   1.0e-6   2.0      3    1       250          500
************************************************************************
************************************************************************
* minor edit requests
************************************************************************
************************************************************************
*
*crdno        variable       parameter
*
301  count    0
302  dt       0
303  dtcrnt   0
304  cputime  0
305  errmax   0
306  emass    0
307  tmass    0
310  mflowj   145010000
311  mflowj   145020000
312  mflowj   716000000
313  mflowj   711000000
314  mflowj   175010000
315  mflowj   175020000
316  tempf    130010000
317  tempf    160010000
318  cntrlvar 1
319  cntrlvar 2
320  testda   2
321  testda   3
322  testda   4
20800001 testda 2
20800002 testda 3
20800003 testda 4
*
*******************************************************************
*******************************************************************
* hydrodynamic components
*******************************************************************
*******************************************************************
1300000   pmpsuca2          pipe   * loop a2 rc pump suction
1300001   1
1300101    4.2761  1
1300301       25.956   1
1300401       0.0   1
```

```
1300601          -90.   1
1300701    -25.956   1
1300801  .00030         0.    1
1301001   00     1
1301201  3   2200.0   550.0      0.0            0.0       0.0       1
*
1450000  clbrcha2        branch
1450001  2         0
1450101  10.0  5.4064       0.     0. -90.0  -5.4064  .00015      0. 00
1450200  3   2200.0    550.0
1451101  160010000 145000000   4.2761   1.0     1.0    0100
1452101  130010000 145000000   4.2761   1.0     1.0    0100
1451201  0.0       0.0        0.0
1452201  0.0       0.0        0.0
*
1600000  pmpsuca1          pipe
1600001  1
1600101   4.2761  1
1600301      25.956   1
1600401       0.0   1
1600601         -90.   1
1600701    -25.956   1
1600801  .00030         0.    1
1601001   00     1
1601201  3  2200.0   550.0   0.0                0.0       0.0       1
*
1750000  clbrcha1        branch
1750001  2         0
*1750101   10.0  5.4064       0.     0. -90.0  -5.4064 .00015      0. 00
1750101  10.0  5.4064       0.     0. -90.0  -5.4064 .01000      0. 00
1750200  3  2200.0   550.0
1751101  175010000 160000000   4.2761   1.0     1.0    0100
1752101  175010000 130000000   4.2761   1.0     1.0    0100
1751201  0.0       0.0        0.0
1752201  0.0       0.0        0.0
*
*
7100000  lpa1hpit       tmdpvol
7100101  1.0e6    10.0 0.0     0.    -90.0  -10.0    0.      0.     00
7100200    3
7100201      0.        2200.0           90.
*
*
7110000  lpa1hpif    tmdpjun
7110101  710010000  175000000        .0246
7110200   1
7110201   0.0    0.0       0.0       0.0
7110202   10.0   96.0      0.0       0.0
*
*
7150000  lpa2hpit       tmdpvol
```

```
7150101   1.0e6   10.0  0.0     0.    -90.0  -10.0    0.      0.     00
7150200     3
7150201       0.        2200.0          550.
*
*
7160000   lpa2hpif    sngljun
7160101   145010000  715000000      10.0    1.0 1.0 0
7160201 0    0.0     0.0       0.0
*
20500100  dtempf  sum  1.0 0.0 1
20500101  0.0  1.0 tempf  160010000  -1.0  tempf  130010000
*
20500200  dtempf  sum  1.0 0.0 1
20500201  0.0  1.0 tempf  130010000  -1.0  tempf  160010000
*
.     * end of input stream
```