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^{TM} LOCA (FSLOCA) Methodology)"

REQUEST FOR ADDITIONAL INFORMATION (RAI)

FOURTH SET OF RAI QUESTIONS

RAI Questions 36 through 45

Table 1: Summary of RAI Questions

Question No.	Subject	Date Issued	Date Responded	Disposition (O/C)[†]	Note
Set 1	**Questions 1 through 19**				
1	WCOBRA/TRAC MOD7A Revision 7				
2	TRAC-PF1/MOD2 Code				
3	Large-break LOCA (LBLOCA) and small-break LOCA (SBLOCA) phenomena identification and ranking tables (PIRTs)				
4	End of blowdown				
5	Gap conductance				
6	Pressurizer response				
7	Long-term cooling and PIRT				
8	SBLOCA boundary and Region-I to Region-II boundary				
9	Worst SBLOCA				
10	Loss-of-offsite power (LOOP) versus RCPs operating				
11	LOOP seal behavior				
12	Worst break sampling				
13	Decay heat multiplier/sampling				
14	Number of SBLOCA cases sampled: 93 versus 124				
15	SBLOCA upper limit break size				
16	Long-term cooling restriction				
17	Swelled or two-phase mixture level versus collapsed level				
18	High pressure safety injection (HPSI) curve basis and uncertainty				
19	SBLOCA axial power shape				
Set 2	**Questions 20 through 29**				
20	^{235}U, ^{238}U, and ^{239}Pu decay heat uncertainty fits to ANS 5.1-1979				
21	^{235}U, ^{238}U, and ^{239}Pu decay heat and uncertainty comparison to ANS 5.1-1979				
22	^{235}U, ^{238}U, and ^{239}Pu decay heat uncertainty comparison to ANS 5.1-1979				
23	Burnup limit in assessing kinetics parameters				
24	Editorial				
25	Utilized codes				
26	Actinides decay heat power				
27	Decay heat in demonstration plant analyses				
28	Decay heat uncertainty distribution				
29	Decay heat sampling approach				

(†) O=Open; C=Closed.

Table 1: Summary of RAI Questions (Continued)

Question No.	Subject	Date Issued	Date Responded	Disposition (O/C)[†]	Note
Set 3	**Questions 30 through 35**				
30	Scaling of the Westinghouse vertical Condensation on Safety Injection (COSI) test facility and tests				
31	Westinghouse vertical COSI downcomer condensation				
32	Westinghouse vertical COSI heat loss				
33	Westinghouse vertical COSI data and condensation outside the jet region				
34	Westinghouse vertical COSI data qualification				
35	Scale impact on cold leg condensation				
Set 4	**Questions 36 through 44**				
36	Fuel thermal conductivity model				
37	Burnup impact on fuel thermal conductivity and initial stored energy				
38	Treatment of fuel burnup dependant parameters				
39	Fuel burnup sampling				
40	Fuel burnup limit in FSLOCA methodology				
41	Nuclear fuel rod special model changes				
42	Nuclear fuel rod special models validation				
43	Dummy rod component models				
44	Fuel rod material properties				
45	Validity of Wilks theorem				

[†] O=Open; C=Closed.

Question #36: Fuel Thermal Conductivity Model

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 11.4, “Thermal Properties of Nuclear Fuel Rod Materials,” explains that the COBRA/TRAC-TF2 default nuclear fuel rod model computes the UO_2 thermal conductivity from a MATPRO-9 correlation to reduce computer time. It is also explained that this correlation has the same error band of 0.2 W/(m-K) and gives very nearly the same conductivity over the expected operating range of 500 K to 3,000 K when compared to the more complex version in MATPRO-11. Section 11.4 also states that an additional optional model is also provided in WCOBRA/TRAC-TF2 to account for the effects of burnup on thermal conductivity. The model, referred to as “the modified Nuclear Fuel Industries (NFI) model,” is described as based on the Nuclear Fuels Industries (NFI) model by Ohira and Itagaki, on pages 541-549 of “Thermal Conductivity Measurements of High Burnup UO_2 Pellet and a Benchmark Calculation of Fuel Center Temperature,” in Proceedings of the ANS international topical meeting on LWR Fuel Performance, Portland, Oregon, March 2-6, 1997. Section 11.4 also provides the range of applicability of the modified NFI correlation with regard to temperature, rod-average burnup and as-fabricated density in accordance with NUREG/CR-6534, “FRAPCON-3 Updates, Including Mixed-Oxide Fuel Properties,” Vol. 4, 2005.

Please clarify the following items related to the default nuclear fuel rod model in WCOBRA/TRAC-TF2 and the modeling approach to account for the effects of burnup on fuel thermal conductivity in LOCA analyses.

(1) The default thermal conductivity model in WCOBRA/TRAC-TF2, based on a MATPRO-9 correlation, does not explicitly account for fuel thermal conductivity degradation with burnup. Please describe the purpose of this model and state the conditions under which its application in LOCA analyses is considered acceptable and justify so. If overestimation of thermal conductivity can be associated with the application of the model for such analyses, are there any other code adjustments in WCOBRA/TRAC-TF2 to compensate for this limitation.

(2) The additional optional model implemented in WCOBRA/TRAC-TF2 to account for the effects of burnup on thermal conductivity is based on a modification of the NFI correlation, which agrees with Equation (2.3-9) in NUREG/CR-7024, “Material Property Correlations: Comparisons between FRAPCON-3.4, FRAPTRAN 1.4, and MATPRO,” March 2011. Such a burnup dependant model was not available in the previous ASTRUM LBLOCA methodology documented in WCAP-16009-P-A (Nissley, M. E., et al., 2005). Please describe the conditions under which this “additional optional model” is considered applicable in FSLOCA LOCA analyses and provide justification.

(3) Please explain how the WCOBRA/TRAC-TF2 nuclear fuel rod model was evaluated for predicting degradation of fuel thermal conductivity with burnup. Describe how contributions from other processes and models such as gap conductance, fission gas release, and radial power profile were taken into consideration in the evaluation. Present analysis results, if available, and provide references to existing assessments that demonstrate the applicability of the model for the purposes of the FSLOCA methodology applications. Include findings from benchmarking against measured data, if available.

(4) The current Westinghouse fuel rod design methodology (approved by NRC in July 2000) is based on the Performance Analysis and Design (PAD) 4.0 fuel performance code, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A/WCAP-15064-NP-A, Revision 1. The PAD 4.0 code has a thermal conductivity model with no burnup dependence. Please explain if specialized fuel performance codes were used in support of the evaluation of the WCOBRA/TRAC-TF2 nuclear fuel rod model for predicting fuel thermal conductivity degradation with burnup. If this was the case, please present the assessment results and include comparison of prediction results for fuel temperatures and rod internal pressures obtained by the codes using the same input conditions. Provide references to the available assessment documentation.

Question #37: Burnup Impact on Fuel Thermal Conductivity and Initial Stored Energy

Concerned about the impact of irradiation on fuel thermal conductivity, the U. S. Nuclear Regulatory Commission (NRC) issued Information Notice (IN) 2009-23, dated October 8, 2009 (ML091550527). In particular, IN 2009-23 states that "safety analyses performed for reactors using pre-1999 methods may be less conservative than previously understood." WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 11.4, "Thermal Properties of Nuclear Fuel Rod Materials," explains that the WCOBRA/TRAC-TF2 nuclear fuel rod model uses a default UO_2 thermal conductivity model based on a MATPRO-9 correlation that does not account for the effect of degradation with burnup. An additional optional model based on the Nuclear Fuels Industries (NFI) model by Ohira and Itagaki (1997) is provided to account for the effects of burnup on thermal conductivity.

With regard to LOCA applications, WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 8.6 explains that the steady state fuel temperature is calibrated against the PAD 4.0 code and refers to WCAP-16996-P Section 29. Subsection 29.4.2.2, "Initial Calibration of the Steady-State Condition for the Nuclear Rods," explains that the initial fuel temperature and rod internal pressure for Westinghouse pressurized water reactors (PWRs) are calibrated against the PAD 4.0 fuel performance code, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P, Revision 1, 1999. The calibration for Combustion Engineering (CE) PWRs is performed against the FATES3B code, "Improvements to Fuel Evaluation Model," CEN-161(B)-P, Supplement 1-P-A, CE, dated 1992. Subsection 29.4.2.2 of WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 29, also states that "the initial fuel temperature is a function of the peak linear heat rate and burnup."

Please clarify the following items related to the nuclear fuel rod model and modeling approach in WCOBRA/TRAC-TF2 with regard to accounting for the effects of fuel burnup in LOCA analyses.

(1) Please explain how the FSLOCA methodology accounts for fuel burnup effects in obtaining core thermal-hydraulic parameters and fuel thermal response under steady state for the purpose of initialization of LOCA analyses. Include consideration of factors related to different reactor fuel cycles, reactor operation time in a cycle, and core nodalization. The FSLOCA methodology core nodalization scheme models a single hot rod and a hot assembly and represents the rest of the core by 3 separate

assembly groupings: (1) low power assemblies on core periphery, (2) average power interior assemblies under guide tube structures, and (3) average power interior assemblies located under other structures. Please explain how WCOBRA/TRAC-TF2 accounts for individual fuel assembly burnup levels for each of the fuel rods that model the reactor core and justify any assumptions.

(2) If results from any other codes are used in the FSLOCA methodology to initialize, calibrate, benchmark, match, or in other way alter WCOBRA/TRAC-TF2 calculated results that have an impact on the initial pellet stored energy, please identify these codes, the frozen code versions used, and their approval status with the NRC. In addition, please document in details and explain such calibrating techniques and describe related algorithms, expressions, criteria, limitations, and assumptions. Justify the applicability and appropriateness of such techniques to account for the effects of fuel thermal conductivity degradation with burnup. Cleary explain how results from the PAD 4.0 or FATES3B codes are used if the code has a thermal conductivity model with no burnup dependence.

(3) If the FSLOCA methodology employs initial calibration of the steady state condition for the nuclear rods by altering the initial fuel temperature and rod internal pressure, please explain why WCOBRA/TRAC-TF2 predictions results for LOCAs should be considered acceptable in terms of describing the core fuel transient responses. In this regard, please present any supporting analyses, if available.

Question #38: Treatment of Fuel Burnup Dependant Parameters

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 29, “Assessment of Uncertainty Elements,” states that “many fuel related parameters are a function of burnup.”

Please clarify the following items related to the consideration of burnup effects on nuclear fuel rod related parameters in the FSLOCA methodology and the accounting for such effects in LOCA analyses.

(1) Please identify the parameters that have been identified as dependent on variability in the fuel burnup. Describe the importance of each parameter with regard to its possible impact on steady state initialization results and on LOCA transient predictions obtained by using the FSLOCA methodology.

(2) Please explain how the functional dependence of each of the identified parameters on the fuel burnup accounts for variability in burnup and provide the burnup range that was considered. Describe how the burnup sampling process proposed for the FSLOCA methodology affects the treatment of each of these parameters.

(3) From the list of parameters identified as burnup dependent, please identify those that are sampled on their own in the FSLOCA methodology. Explain how burnup is accounted for in the definition of the sampling ranges and sampling distributions for each of these burnup dependant parameters.

(4) Please provide a table that summarizes the findings in response to the above identified items.

Question #39: Fuel Burnup Sampling

Please describe the process of fuel burnup sampling as proposed in the FSLOCA methodology. Describe the sampling technique and explain how the proposed sampling approach accounts for fuel burnup variability with regard to space in consideration of fuel assemblies with different burnup in the core (e.g., fresh, once-burned, and twice-burned fuel) as well as for fuel burnup variability with regard to time in consideration of different fuel cycles (non-equilibrium and equilibrium) and reactor operation time in a cycle. In addition to the information in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Subsection 29.4.1.1, “Time in Cycle,” please explain if any such aspects due to burnup variability in space and time have been simplified or ignored in the proposed FSLOCA methodology fuel burnup sampling approach and provide justification.

Question #40: Fuel Burnup Limit in FSLOCA Methodology

Please clarify if the FSLOCA methodology is limited to 62,000 megawatt-days per metric ton of uranium rod average burnup for LOCA licensing applications. If so, please explain how this burnup limit was established and where it is described in the FSLOCA methodology. In addition, please explain if the rod average burnup limit has been adequately and consistently accounted for in the proposed FSLOCA sampling methodology.

Question #41: Nuclear Fuel Rod Special Model Changes

The nuclear fuel rod conductor in the FSLOCA methodology has several special models used for analyses of nuclear fuel rods. These models include a fuel rod quench front model, a dynamic pellet-cladding gap conductance model, a fuel rod deformation model, and a cladding reaction model. These models are described in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 8.3.1, Section 8.3.2, Section 8.4, and Section 8.5, correspondingly.

The previous ASTRUM LBLOCA Evaluation Model (EM) documented in WCAP-16009-P-A (Nissley, M. E., et al., 2005), describes the fuel rod quench front model in Section 7-3-1, the dynamic pellet-cladding gap conductance model in Section 7-3-2, the fuel rod deformation model in Section 7-4, and the cladding reaction model in Section 7-5.

Please clarify the following items related to the above identified models implemented in the fuel rod conductor in WCOBRA/TRAC-TF2.

(1) The fuel rod quench front model in the ASTRUM FSLOCA EM is presented in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 8.3.1, which contains Equations (8-25a) and (8-25a). The same model for the 2005 ASTRUM LBLOCA EM is presented in Section 7-3-1 of WCAP-16009-P-A, which contains Equation (7-25). Please identify and describe any changes introduced to the FSLOCA methodology model in comparison to the 2005 ASTRUM LBLOCA model.

(2) The dynamic pellet-cladding gap conductance model in the ASTRUM FSLOCA EM is presented in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 8.3.2, which contains Equations (8-26) through (8-35). The same model for

the 2005 ASTRUM LBLOCA EM is presented in Section 7-3-2 of WCAP-16009-P-A, which contains Equations (7-26) through (7-35). Although the models appear identical, please identify and describe any changes introduced to the FSLOCA methodology model in comparison to the 2005 ASTRUM LBLOCA model.

(3) The fuel rod deformation model in the ASTRUM FSLOCA EM is presented in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 8.4, which contains Equations (8-36) through (8-73). The same model for the 2005 ASTRUM LBLOCA EM is presented in Section 7-4 of WCAP-16009-P-A, which contains Equations (7-36) through (7-72). Some differences between both models were identified. Thus, the description in Section 8.4.2, "Effects of Fuel Rod Deformation on Core Thermal-Hydraulics," of WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, appears different in its last Subsection "Flow Blockage Due to Rod Deformation," which contains Equations (8-72) and (8-73) for the blockage form loss. The corresponding Section 7-4-2 ends with a Subsection "Continuity and Momentum Cell Flow Areas," which includes only Equation (7-72) for the flow area reduction factor. At the same time, Equation (7-69) for the outer radius of the heat transfer node containing the burst elevation in Section 7-4-2 is omitted in Section 8.4.2. Please identify and describe any changes introduced to the FSLOCA model in comparison to the 2005 ASTRUM LBLOCA model and present the validation base for the introduced modifications.

(4) The cladding reaction model in the ASTRUM FSLOCA EM is presented in Section 8.5 of WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, which contains Equations (8-74) through (8-85). The same model for the 2005 ASTRUM LBLOCA EM is presented in WCAP-16009-P-A Section 7-5, which contains Equations (7-73) through (7-87). Both model descriptions appear identical with the exception that the 2005 ASTRUM LBLOCA EM includes a description of the oxidation kinetics of ZIRLO™ cladding material manufactured by Westinghouse at high temperatures based on a model by D. L. Burman, "ZIRLO™ High Temperature Oxidation Tests," Appendix E to WCAP-12610, Westinghouse, Pittsburgh, PA, 1990. This model description along with the pertaining Equations (7-80) through (7-82) is not provided for the FSLOCA model. Please identify and describe any changes introduced to the FSLOCA model in comparison to the 2005 ASTRUM LBLOCA model and present the validation base for the introduced modifications.

Question #42: Nuclear Fuel Rod Special Model Validation

The special models of the nuclear fuel rod conductor implemented in the FSLOCA methodology to analyze the nuclear fuel rods behavior include the fuel rod quench front model, dynamic pellet-cladding gap conductance model, fuel rod deformation model, and cladding reaction model. These models are described in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 8.3.1, Section 8.3.2, Section 8.4, and Section 8.5, correspondingly. As explained in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Subsection 29.4.2.2, "Initial Calibration of the Steady-State Condition for the Nuclear Rods," specialized fuel performance codes are used by Westinghouse for stored energy and rod pressure inputs to LOCA analyses performed with the FSLOCA methodology.

Please clarify the following items related to the special models of the nuclear fuel rod conductor implemented in WCOBRA/TRAC-TF2 for describing the behavior of nuclear fuel rods.

(1) Please explain if some of the nuclear fuel rod special models or major modeling features in these models have been previously reviewed by the NRC. In particular, clarify if this has been the case as part of the review of other fuel rod performance codes used by Westinghouse for LOCA analyses. If this is the case, please identify these codes, code versions, models and/or modeling features that apply to the WCOBRA/TRAC-TF2 nuclear fuel rod conductor model, provide the review outcome, and summarize major relevant review findings and conclusions. Please provide references for the safety evaluations by the staff.

(2) Please explain if specialized fuel performance codes were used as part of the evaluation of the WCOBRA/TRAC-TF2 nuclear fuel rod special models. In such a case, please present the assessment results and include comparison of prediction results for governing parameters obtained by the codes using the same input conditions. Provide references to the available assessment documentation.

Question #43: Dummy Rod Component Models

As a major difference between the FSLOCA methodology and the previous ASTRUM LBLOCA Evaluation Model (EM) documented in WCAP-16009-P-A (Nissley, M. E., et al., 2005), WCOBRA/TRAC-TF2 features a dummy rod component, **[**

] WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 8.6 describes the dummy rod component.

Please clarify the following items related to the new dummy rod component implemented in WCOBRA/TRAC-TF2.

(1) Please explain if the dummy rod component models are consistent with the corresponding models in WCOBRA/TRAC-TF2. If corresponding models are not fully consistent, please identify only the differences and explain why such differing features in the dummy rod model are considered appropriate and valid for the intended functions of the dummy rod component.

(2) Please identify the models in the dummy rod component that have no counterpart models in WCOBRA/TRAC-TF2. For each such model, please explain if the model or major modeling features in the model have been previously reviewed by the NRC. If this is the case, please identify these codes, code versions, models and/or modeling features that apply to the WCOBRA/TRAC-TF2 dummy rod component, provide the review outcome, and summarize major relevant review findings and conclusions. Please provide references for the safety evaluations by the staff.

(3) Please explain how the dummy fuel rod model in WCOBRA/TRAC-TF2 was evaluated. Present analysis results, if available, and provide references to existing assessments that demonstrate the applicability of the model for the purposes of LOCA analyses performed with the FSLOCA methodology. Please explain if the model was benchmarked against any available tests.

Question #44: Fuel Rod Material Properties

WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Section 8.2, "Conductor Geometries Modeled in the Vessel," explains that the nuclear fuel rod model requires minimum user input and uses material properties specified by input or "defaulted to uranium-dioxide and zircaloy." The default properties are calculated using correlations from MATPRO-Version 11 (Revision 1) as documented in NUREG/CR-0497, Revision 1, 1980.

Subsection 8.3.2, "Pellet-Cladding Gap Conductance Model," of the Section 8.3, "Fuel Rod Modeling," states that the material property correlations in the simplified dynamic gap conductance model were taken exclusively from MATPRO-11 (Revision 1). In computing radiant heat transfer, the fuel emissivity and that of the cladding inner surface are based on data from MATPRO-11 (Revision 0). The gas mixture conductivity is determined from the conductivities of the constituent gases using a simplified version of the model in MATPRO-11 subroutine GTHCON. The conductivities of helium, xenon, argon, krypton, hydrogen, and nitrogen gases are calculated using correlations from MATPRO-11 (Revision 1). The interfacial pressure for the pellet-cladding contact conductance is calculated with the fuel rod deformation model and is non-dimensionalized using the Meyer hardness calculated from MATPRO-11 (Revision 1).

Subsection 8.4.1, "Deformation Mechanisms Fuel Pellet Thermal Expansion," of the Section 8.4, "Fuel Rod Deformation Model," explains that the axial and radial thermal expansion of the fuel is calculated using a MATPRO-11 (Revision 1) correlation for thermally induced strain in UO_2. The correlation was simplified by omitting the corrections for molten fuel and mixed oxide. The axial and radial thermal expansion of the cladding is also calculated using correlations from MATPRO-11 (Revision 1).

The most recent version of the material properties library MATPRO is documented in NUREG/CR-6150, Vol. 4, Rev. 2, "SCDAP/RELAP5/MOD 3.3 Code Manual: MATPRO - A Library of Materials Properties for Light-Water-Reactor Accident Analysis," 2001. A comparative study between the current versions of FRAPCON-3 and FRAPTRAN, which use a relatively consistent set of correlations for applied properties, and the latest MATPRO properties is documented in NUREG/CR-7024, "Material Property Correlations: Comparisons between FRAPCON-3.4, FRAPTRAN 1.4, and MATPRO," 2011. In addition to comparing various correlations, correlation-to-data comparisons for FRAPCON-3, FRAPTRAN, and MATPRO properties are also provided in NUREG/CR-7024.

Please explain how material properties in the WCOBRA/TRAC-TF2 nuclear fuel rod model described in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, Sections 8.2, 8.3, and 8.4 compare with the latest MATPRO library and data provided in NUREG/CR-6150 (Siefken et al., 2001) and NUREG/CR-7024 (Luscher and Geelhood, 2011). Please clarify which cladding materials can be modeled using the default material properties for zircaloy in the WCOBRA/TRAC-TF2 nuclear fuel rod model.

Question #45: Validity of Wilks theorem

Please demonstrate that the assumption for the validity of Wilks theorem holds with regard to the application of the code described in WCAP-16996-P/WCAP-16996-NP, Volumes I, II, and III, Revision 0, in quantifying a single probabilistic statement of safety for the full spectrum of breaks, the full spectrum of model parameters and their variation, and the models of the engineered safety systems for small, intermediate and large break LOCAs. That is, there are no disjoint density functions of the figures of merit, or you can identify them and take them into account in the application of Wilks theorem.