ENCLOSURE 2

MFN 16-008

Response to Request for Additional Information Regarding Review of Licensing Topical Reports NEDE-33005P and NEDO-33005, "Licensing Topical Report TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant-Accident Analyses for BWR/2-6"

Non-Proprietary Information – Class I (Public)

IMPORTANT NOTICE

This is a non-proprietary version of Enclosure 1, from which the proprietary information has been removed. Portions of the enclosure that have been removed are indicated by an open and closed bracket as shown here [[ ]].

SNPB RAI-67

Licensing Topical Report (LTR) Figure 5.2-8 appears to be an inadvertent repetition of Figure 5.2-5. Please provide the corrected figure.

RAI-67 Response

Figure 5.2-8 will be replaced with the correct one in the LTR. It was also noted that Figure 5.2-5 is missing the "Theta Nodalization" legend to denote the azimuthal nodalization sensitivity results so this missing legend will be added to Figure 5.2-5.

LTR Impact

The following figure is to replace LTR Figure 5.2-8:

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The following figure is to replace LTR Figure 5.2-5:

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SNPB RAI-68

Recent staff experience, in combination with the previous response to RAI 8 (regarding the treatment of evaluation model (EM) errors and changes) suggests that additional information is required concerning the treatment of input changes, plant modifications, and code changes. The treatment requires discussion both in the context of estimating the effect of an error or change, and in the context of performing a reanalysis, whether in fulfillment of a Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46(a)(3)(ii) commitment or statement to “include with the report a proposed schedule for providing a reanalysis,” or simply for the purpose of developing a new baseline to eliminate an extensive error/change rackup list, or to analyze a major plant change like a fuel design change or operating domain extension. Discuss, or provide applicable procedures or modeling guidance that explains, how to discern between changes that may be estimated using engineering principles, using revision analysis (such as analyzing the effect of a change using the same population sample), or using more comprehensive techniques, such as generating a new sample or re-analyzing the break spectrum/operating domain.

RAI-68 Response

The reporting requirements mandated by 10 CFR 50.46 impose on the licensee for action. As the methodology owner, the fuel vendor supports the evaluations in case there are changes and/or errors in the evaluation model. The 10 CFR 50.46 reporting process is beyond the scope of approval this LTR is seeking. It would be expected that with an approved TRACG LOCA evaluation model, the resolution of changes and errors subsequently identified would follow a similar process as currently employed. The discussion below is provided as clarifying commentary.

According to 10 CFR 50.46(a)(3)(ii), reporting is required for a change in an evaluation model, a discovered error in an evaluation model, or an error in a plant-specific application of the evaluation model. This would not change for the TRACG LOCA methodology. Reporting for significance and compliance are checked for those changes or errors that affect the calculated peak cladding temperature (PCT). For these, the effect on the limiting emergency core cooling system (ECCS) analysis must be estimated.

In estimation of the PCT effect, engineering judgment is rarely used. Such approach would be only applicable if the effect of the change can be explicitly defined and the effect is conclusively confined. For example, in case of a discovered error, if the correction results in a negligible (less than 0.1%) change in critical parameters, then it would be possible to argue a negligible effect (i.e., 0°F). Also, some changes to the computer code, such as computing platform change or recompilation can be estimated as a 0°F effect using engineering judgment if the results after the change show no particular trend and one-to-one comparisons show fairly small difference in computed PCTs.

In some cases, it would be possible to use known sensitivities to estimate the effect. One example [[

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In some other cases, an estimate based on first principles can be made if the change or error correction provides a justifiable means for this approach. For example, if the heat transfer coefficient would be 10% lower when an error is corrected, it would be reasonable to estimate an increase in computed temperatures by about 10% of the temperature difference where the heat flows. In such instances, it would be prudent to take into account other non-linear effects. Such approximation would have limited applicability, in other words, would be acceptable if the resulting effect is relatively small.

The process employed by GE Hitachi Nuclear Energy (GEH) [[

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It would be expected that an accumulation of several changes or errors of small effect could be reported and monitored via the (annual) reporting cycle of the regulation as is done currently and not require specific re-analysis until the sum of the absolute values of these would reach the significance standard (50ºF) of the regulation. Likewise, any error that resulted in a plant not conforming to acceptance criteria would require immediate remediating action to return to compliance along with subsequent re-analysis, those provisions of the regulation being operative with TRACG analyses in the same way as currently in force. Just as it is understood for the current process, an error found in the evaluation model is interpreted as a condition at variance with the evaluation model as approved by the Nuclear Regulatory Commission (NRC). Resolution of an error, with its resulting reporting of the effect will have the salutary effect of bringing the plant analysis back into conformance with the evaluation model of the LTR as intended to have been approved via the Safety Evaluation Report (SER).

As discussed in Chapter 9 of the LTR, there are some changes to the analyses that cannot be characterized as a 10 CFR 50.46 change. These include a new fuel introduction or a significant change in plant's ECCS configuration including flow capacities. In the case of a fuel type change, it would be reasonable to [[

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In any of the examples delineated above, [[

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Related to the 10 CFR 50.46 reporting topic, some changes in TRACG LOCA EM since it was submitted for review in 2011 (as of this writing more than 4 years ago) are acknowledged. Because there is no United States (US) plant that relies on TRACG LOCA methodology for 10 CFR 50.46 compliance, these changes need not be reported as part of a plant application. They will be resolved by the initial application analysis. Errors or changes to the EM discovered after it is approved by NRC will then be subject to reporting according to 10 CFR 50.46 requirements.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-69

The uncertainty analysis appears to be based on correlations being used only within their applicable limits. Please explain what code features or processes ensure that correlations are used within applicable limits. For example: Does the code flag if correlations are used outside of their range of applicability? Are correlation ranges of applicability checked and validated by the analyst as part of the calculation process, or by a reviewer as part of the quality assurance process?

RAI-69 Response

Reference R69-1 provides the technical basis for the correlations used as auxiliary relations for closure of the basic mass, energy, and momentum conservation equations. The correlations are grouped into two broad categories: (1) those related to determination of the flow regime; and (2) those related to interfacial processes and wall heat transfer. Use of specific correlations within their applicable limits is largely achieved within TRACG by choosing correlations that have been demonstrated to be generically applicable over a wide application range for a particular flow regime or particular heat transfer mode and making a smooth transition to another correlation as the boundary for the range of application is approached. Details of how this process has been implemented in the code and qualified are provided in Section 5 and Section 6 of Reference R69-1. The remaining paragraphs of this response will highlight and summarize relevant key aspects of the code design, qualification of the models as coded against data, and describe the processes that regulate the use of the code so that the models are applied appropriately.

As noted in the introduction of Section 5 of Reference R69-1, the constitutive correlations expressing the rates of exchange of mass, momentum, and energy between each phase and their surroundings depend on the flow patterns. Thus the flow regime in each hydraulic cell is needed before proceeding with the solution of the flow equations for that cell. [[

]] The liquid-continuous regime applies to single-phase liquid flow, bubbly/churn flow, and inverted annular flow. The vapor-continuous regime applies to annular flow, dispersed droplet flow, and single-phase vapor flow. The correlations used to derive the expressions describing the relationship between the vapor and liquid velocities are specific and applicable to the specific regime where they are used as described in Section 5.1 of Reference R69-1. A transition regime between the liquid-continuous regime and the vapor-continuous regime is defined in terms of a transition void fraction that is calculated according to Equation (5.1-6) based on flow conditions and geometry. This process of a dynamic transition regime is what facilitates the use of the simple flow regime map, ensures appropriate application of correlations describing interfacial shear, and produces accurate predictions of void fractions over the entire range from 0.0 to 1.0. The accuracy of the calculated void fractions is demonstrated by the separate-effects qualification cases documented in Section 3.1 of Reference R69-2.

Within the vapor-continuous regime annular flow may be accompanied by a varying amount of dispersed droplets as determined by the entrainment model described in Section 5.1.2 of Reference R69-1. The limited experimental range of pressures used for development of Ishii's correlation is addressed in part via the use of dimensionless correlation parameters that account for the relationships between fluid properties. For higher pressures approaching boiling water reactor (BWR) operating pressures, the entrainment correlation has been indirectly validated through comparisons to void fraction data that cover the needed application range as described in Section 6.1.8 of Reference R69-1. Specifically for LOCA applications, qualification of the entrainment and interfacial shear models at intermediate pressures has been added by comparisons to Toshiba void fraction data as indicated in Section 3.1.6 of Reference R69-2.

Section 6 of Reference R69-1 describes the constitutive correlations for interfacial shear and heat transfer, wall friction, and heat transfer for the individual flow patterns determined using the models from Section 5 that are summarized above. The models and correlations in Section 6 of Reference R69-1 are intended to cover the application range for a wide variety of BWR transients and LOCAs. The intended range of applicability for the various BWR regions in the reactor vessel is shown in Table 6-1 of Reference R69-1. Comparisons to the Table 6-1 intended application ranges are provided in the subsections of Section 6 for each of four model groups: (1) interfacial shear, (2) pressure drop, (3) interfacial heat transfer, and (4) wall shear. For each model group a table summarizes the suitability of the models in that group to predict the phenomena in the intended ranges given for each component in Table 6-1. Table 6-2 summarizes the applicability of the interfacial shear models for predicting the interfacial shear phenomena; Table 6-3 for pressure drop; Table 6-10 for interfacial heat transfer; and Table 6-18 for wall heat transfer. These four tables are supported by specific subsections within Section 6 of Reference R69-1 that address the applicability of specific correlations. Examples for each of the four model groups are provided below.

Interfacial shear models and correlations are described in Section 6.1 of Reference R69-1. Section 6.1.3.3 addresses the applicability for the correlations for the interfacial shear for bubbly flow; Section 6.1.4.3 for annular flow; Section 6.1.5.3 for droplet flow; Section 6.1.6.3 for annular/droplet flow; and Section 6.1.7.4 for counter-current flow. Ranges of application are specifically addressed. As described in the second paragraph of this response, use of the flow regime logic establishes where a specific correlation gets used to prevent for example use of a correlation for bubbly/churn flow in a flow regime where annular flow is indicated or vice versa. The models as coded are assessed by comparisons to data that are either presented in Reference R69-1 or referenced to Reference R69-2. The bulk of the data available for the evaluation of the interfacial shear and the wall friction are void fraction and pressure drop data as indicated in Section 6.1.1 of Reference R69-1. The quantification of model biases and uncertainties needed for statistical analyses for LOCA applications were provided in the LOCA Application LTR determined from comparisons to data. These are the assessments that support the overall assessment of the interfacial shear model summarized in Section 6.1.8 of Reference R69-1 and the summary conclusions of Table 6-2 range-of-application coverage relative to the intended (or needed) ranges in Table 6-1.

Models and correlations for wall friction and form losses are described in Section 6.2 of Reference R69-1. The basic models and correlations used for wall friction in Section 6.2.1 are common ones that have been well established and extensively tested. Extensive comparison to rod bundle pressure drop data has been used to qualify modifications to the Chisholm correlation to achieve better two-phase performance in the fuel channel. Details for the code implementation are given in Section 6.2.1.4 and the applicability of the models and correlations to single-phase and two-phase conditions is assessed in Section 6.2.1.5.

Models and correlations for form losses are described in Section 6.2.2 of Reference R69-1. A local loss coefficient is input for each flow direction based on the geometry of the flow. User input errors are minimized by using standard templates and automation that calculates the inputs based on geometry recorded and verified in a database. The code also provides an input check that can be activated to assess the input local losses and print a warning for those that do not appear to be consistent with the geometry. Two-phase local loss values are internally calculated using a two-phase multiplier to the single-phase inputs as described in Section 6.2.2.3 of Reference R69-1. Applicability of the two-phase model is based on pressure drop data as described in Section 6.2.2.5 of Reference R69-1.

Overall assessment and applicability of the models for wall friction and form losses is documented in Section 6.2.3 of Reference R69-1. Citations are made to the qualification cases in Reference R69-2. [[ ]] These are the assessments that support the summary conclusions of Table 6-3 of Reference R69-1regarding the range-of-application coverage relative to the intended (or needed) ranges defined in Table 6-1 of Reference R69-1.

The critical flow model and correlations are described in Section 6.3 of Reference R69-1. As indicated in Table 6-7 of Reference R69-1, the code logic prescribes how the different models are integrated so that models specific to a particular range are not misapplied. Applicability of the models as coded is addressed in Sections 6.3.5 and 6.3.6 of Reference R69-1. Qualification using critical flow separate effects data is documented in Section 3.4 of Reference R69-2. There are also many break flow studies [[ ]] that are summarized in Section 6.3.6 of Reference R69-1 and documented in detail in Section 5 of Reference R69-2.

Models and correlations for interfacial heat transfer are described in Section 6.5 of Reference R69-1. Determination of both the total heat exchange and mass transfer rates at the liquid-vapor interface require knowing the interfacial area per unit volume. This interfacial area is determined using the same flow regime map as used for the interfacial shear. Entrained drops in the vapor-continuous phase have a large effect on the interfacial area as do vapor bubbles in the liquid-continuous phase. In addition, the interfacial heat transfer coefficients themselves also depend on the characterization of the flow regime so that correlations are not applied outside the characterization for which they were designed and intended. These characterizations in

Reference R69-1 are organized in the following sections of this reference: 6.5.3 bubbly/churn flow; 6.5.4 annular flow; 6.5.5 droplet flow; 6.5.6 annular/droplet flow; and 6.5.7 transition to annular flow. Each characterization includes a discussion of the applicability of the model as-coded and/or correlation applicability.

As indicated in Section 6.5.11 of Reference R69-1, separate effects assessment of all the models and correlations integrated into the interfacial heat transfer model is not possible. Instead overall assessment is performed by selecting a set of steady state and transient qualification cases with a strong dependency on the interfacial heat transfer. These assessments include examples of sub-cooled boiling and film boiling where interfacial heat transfer effects are significant. The details in Section 6.5.11 of Reference R69-1 are quite lengthy and will not be repeated here. It is worth pointing out that based on combining the theoretical basis and applicability ranges of the individual models, Section 6.5.11 of Reference R69-1 defines for the most important bubbly/churn flow and annular flow regimes specific applicability ranges in terms of pressure, flow rate, dimension, and void fraction. Based on these, it is concluded as summarized in Table 6-10 of Reference R69-1 that the range-of-application coverage for the interfacial heat transfer models is sufficient as measured relative to the intended (or needed) ranges defined in Table 6-1 of Reference R69-1. These models are summarized here in Table R69-1 together with the application ranges.

Table R69-1 Summary of Wall Heat Transfer Models

Model LTR Section(s)	Heat Transfer Mode	Temperature and Fluid Conditions	[[			]]
6.6.3	single phase liquid	[[				
6.6.4	subcooled boiling TL<Tsat					
	nucleate boiling					
						]]

Model LTR Section(s)	Heat Transfer Mode	Temperature and Fluid Conditions	[[			]]
6.6.5	single phase vapor	[[				
6.6.6	determination for onset of boiling transition (dryout)					
						]]

Model LTR Section(s)	**Heat Transfer Mode**	**Temperature and Fluid Conditions**	[[			]]
6.6.7	determination of Tmin	[[				
6.6.8	Transition Boiling					
6.6.9	Film boiling - low voids					
6.6.10	Film boiling - high voids					]]

Model LTR Section(s)	**Heat Transfer Mode**	**Temperature and Fluid Conditions**	[[			]]
6.6.11	Condensation (includes effects of NCGs)	[[				
6.6.12	Thermal Radiation					
6.6.13	Quenching					
6.6.14	Metal-Water reaction					]]

Symbols used in Table R69-1 above:

Symbol	**Description**	**Symbol**	**Description**
{..}	Abbreviation for another correlation in the table	h	Fluid enthalpy
[#]	Reference as cited in R69-1	G	Mass flux
Tw	Temperature at wall surface	P	Pressure of fluid (total)
TL	Temperature of liquid	C_p	Specific heat at constant pressure
TV	Temperature of vapor	Pr	Prandtl number
Tsat	Temperature of saturated water at the steam partial pressure	Re	Mixture Re number
TCHF	Temperature at Critical Heat Flux	ReL	Liquid Re number
Tmin	Minimum Temperature for stable film boiling	xe	Equilibrium quality of fluid
α	Void fraction of fluid	xcrit	GEXL critical quality
q	Heat flux	ρ	Density
β	Non-dimensional group defined by $\beta = \sqrt{(k\rho C_p)_\ell} \Big/ \sqrt{(k\rho C_p)_w}$	k	Thermal conductivity

The code selection logic is used to choose the heat transfer model appropriate for the wall surface temperature and the fluid and flow conditions. Special attention is given to ensure continuous transitions between models at the appropriate conditions. Consider for example the case of single phase liquid heat transfer. The transition between laminar and turbulent flow nominally occurs for Re~2300 but this does not assure that the heat transfer correlations will provide exactly the same value for the heat transfer coefficient (HTC) at this flow condition. A similar issue exists for the transition between the forced and natural convection correlations. Because the application ranges for the correlations overlap, the code logic is to pick the largest heat transfer coefficients predicted by the turbulent, laminar and natural convection correlations. This approach takes advantage of knowledge about the flow dependency in the Dittus-Boelter correlation for turbulent flow that causes it to appropriately yield the highest HTC value of the three at the higher mass fluxes yet appropriately fall under the other two correlations for the lower mass fluxes corresponding to natural circulation and laminar flow. Figure R69-1 illustrates the selection of the highest HTC value from the relevant correlations for single-phase liquid evaluated for a specific fluid state that allows for comparison to the data presented in the cited figures from Reference R69-1. [[

]] Of course, the benefit of expressing the correlations in non-dimensional form is that the correlations scale for other fluid conditions. Note that the HTC as used by the code is determined from the maximum of the Nusselt numbers from the correlations as represented by the solid lines in Figure R69-1. It is evident from the figure that the modeled HTC value determined in this way either agrees well or is conservatively slightly under the data. Additional explanation and justification is provided in Section 6.6.3.3 of Reference R69-1.

As described above, TRACG code design specifically addresses how models and correlations are applied in order to preclude misapplications outside the qualified ranges of application. In some cases modifications have been made and/or additional qualification provided to extend the originally-designed ranges of application. Transition between different models and/or correlations has been designed into the code to address the limited application ranges for some specific models or correlations. Validation of this approach is supported by extensive qualification of the as-coded models to a wide range of test data.

Another code design feature is that TRACG provides checks to enforce the code limitations related to the domain covered by the properties for water, UO_2, and Zr. These property ranges shown in Table R69-2 are quite wide to accommodate many uses of the code. An error message is triggered every time one of these limits is violated and must by procedure be adequately addressed by the responsible engineer to the satisfaction of the independent verifier and responsible manager. In many cases, a fatal abort is triggered for conditions where it is not prudent for the calculation to continue and the resulting message indicates the reason for the abort.

Due to other considerations, the application range for most problems is much more restrictive than the limits in Table R69-2. For example, although Zr properties extend to [[ ]], it is not credible that the fuel rod geometry in the core will be maintained above about 2500 K and for

LOCA licensing applications any evaluated Zr cladding temperature above 1477 K (2200ºF) would not be acceptable. Limitations of this type are addressed administratively by the code Application Statement (contained in Reference R69-3) and are also factored into the controlled automation scripts that are developed for each approved licensing application. Usually technical design reviews are also extensively used to insure applicability of the code when developing a new application that challenges the code in ranges outside where it has been previously applied and proven.

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Figure R69-1 Illustration of Selection of Highest HTC from Single-Phase Liquid Correlations

Table R69-2 TRACG Property Limitations

Minimum Value	Quantity [Symbol]	Maximum Value	Units
[[			
			]]

The TRACG code also provides for rigorous input checking that produces informative, warning, and fatal messages as describe in Reference R69-3. These checks enforce the code limitations for material properties, problem size, and also check for self-consistency in the geometric definitions of the problem and use of applicable correlations depending on the type of physical component that is being represented and simulated. A specific choice of correlation for cases where more than one choice is available is guided by recommendations in the User's Manual and further enforced internally by the code via pre-selection of appropriate default inputs. Most applications employ automated scripts for defining inputs and running the code in order to minimize user errors. These automation scripts have been tested, verified, and are controlled in accordance with our quality assurance program.

The TRACG code provides extensive textual and graphical output. Information contained in the output allows users of the code to determine what models the code is applying as a function of time. Graphical output is especially useful for aligning trends and inflections in the transient responses to allow visualization of interactions between dynamic physical phenomena. Traditional time plots are often used together with animation to present information so that it can be understood and communicated. Users can compare to prior results from a similar problem and can also compare to the data provided in the extensive qualification bases. Both performers and reviewers use these tools and processes to evaluate the appropriateness of the calculated results before they are released as part of a quality record.

References

R69-1 GE Hitachi Nuclear Energy, "TRACG Model Description," NEDE-32176P, Revision 4, January 2008.

R69-2 GE Hitachi Nuclear Energy, "TRACG Qualification," NEDE-32177P, Revision 3, August 2007.

R69-3 GE Hitachi Nuclear Energy, "TRACG04P User's Manual," December 2011.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-70

Given the importance of the low pressure core spray (LPCS) system relative to the limiting peak cladding temperature (PCT) loss-of-coolant accident (LOCA) analyses, provide additional justification of the adequacy of the existing suite of LPCS benchmark tests included in the TRACG benchmark data in Table 4.4-1 of the Qualification Report, addressing the following specific topics:

a. Explain how the database addresses system variability, such as varying nozzle designs.
b. Is the channel power distribution in the testing bounding relative to the BWR/2 design?
c. Explain how the uncertainty from 6x6 through 8x8 fuel data scales to modern 10x10 fuels.
d. Explain how the uncertainty and model corrections discussed in the Qualification Report are applied in the emergency core cooling system (ECCS) EM.

RAI-70 Response

The General Electric (GE) core spray distribution methodology is described in Reference R70-1. The methodology was developed under the key assumption that the condensation (thermodynamic) effects in a steam environment can be handled independently from the hydrodynamic effects of multiple nozzle interactions. Under this assumption, the thermodynamic effects are evaluated from single nozzle spray distribution tests in a simulated reactor steam environment. The hydrodynamic effects of multiple nozzle interactions are determined from single nozzle and full-scale sparger spray distribution tests in an air environment using simulator nozzles. Simulator nozzles are specifically developed spray nozzles which produce spray patterns in an air environment similar to corresponding reactor nozzle spray patterns in a steam environment. Single nozzle tests in steam are performed at 29.5 psia where the steam density is approximately equal to the density of atmospheric air. This maintains dynamic similarity between performance in air and steam.

The GE Steam Sector Test Facility (SSTF) tests at Lynn, Massachusetts, utilizing the BWR/6[1] design (Reference R70-1), confirmed the capability of the methodology to predict spray distribution performance in a steam environment. The results substantiate the key assumption of separability of thermodynamic and hydrodynamic effects. The methodology was approved by the NRC in Reference R70-1.

It is important to note that the purpose of those core spray distribution tests in Reference R70-1 (also in References R70-3 and R70-4) was to determine core spray liquid available at each core location at zero or low steam updraft flow conditions. The core spray distribution generated using GE core spray distribution methodology, therefore, does not depend on specific bundle designs and bundle power.

For the demonstration calculations in the LTR for BWR/2 to 6, [[
]], as discussed in Section 4.4.4 of Reference R70-2 and in the response to Item d below.

[1] Additional SSTF tests were performed utilizing the BWR/4 and 5 sparger design (Reference R70-3).

a. The complete list of test matrix and test results is presented in NEDO-24712-A (Reference R70-1) Appendix C. Among all 37 tests in Reference R70-1, [[

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SSTF tests were initially performed for BWR/6-218, and later extended to BWR/4 and 5-218 by providing BWR/4 and 5 specific nozzle and sparger designs (Reference R70-1 and Reference R70-3). The GE core spray methodology was reviewed and approved by the NRC in Reference R70-1.

b. The application of the GE core spray methodology to a BWR/2 plant (Nine Mile Point Unit 1 (NMP1) in this case) was made in Reference R70-4. This BWR/2 core spray design was compared to BWR/4 and 5 and BWR/6 core spray designs. It was found that the nozzle placement configuration of the BWR/4 and 5 designs is similar to the BWR/2 design, while the nozzle types of BWR/6 are similar to the BWR/2 nozzles (See Table A-1 in Reference R70-4). Core spray tests in both steam and air, according to the requirement of the GE core spray methodology in Reference R70-1, were made and the core spray distribution applicable to this specific BWR/2 were generated (See the results in Figures A-31 and A-32 in Reference R70-4). Section 3.2.4 of Reference R70-4 described how the GE core spray methodology in

Reference R70-1 was applied to NMP1. SSTF tests in steam environment were carried out to confirm the methodology (Reference R70-4 Section 3.2.5). The system pressure effect on the spray was considered (Section A.3). The dual sparger performance was also considered (Section A.6 of Reference R70-4), similar to the study for BWR/4&5 in Reference R70-3.

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[2] This facility can also be used for single nozzle tests.

Although the above discussion is for a specific BWR/2, it is also applicable to other BWR types regarding core spray distribution tests (References R70-1 and R70-3).

c. [[

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References

R70-1 General Electric, "Core Spray Design Methodology Confirmation Tests," NEDO-24712-A, March 1983.

R70-2 GE Hitachi Nuclear Energy, "TRACG Qualification," NEDE-32177P, Revision 3, August 2007.

R70-3 NUREG/CR-1707, "BWR Refill-Reflood Program Task 4.2 – Core Spray Distribution Final Report," September 1980.

R70-4 General Electric, "Performance Evaluation of the Nine Mile Point Unit 1 Core Spray Sparger," NEDE-30241, September 1983.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-71

Please clarify the methodology for validating the acceptability of changes to the generic nodalization in the LTR for plant-specific calculations (as discussed in Section 5.2 of LTR), and justify its sensitivity for distinguishing the potential for nodalization changes to affect the determination of 95/95 upper tolerance limits for assessing compliance with the criteria of 10 CFR 50.46.

RAI-71 Response

The generic nodalization is representative and typical of the least-detailed nodalization that is considered acceptable for most applications. Additional details may be added or changed from the generic nodalization for specific applications provided the effect on modeling biases and uncertainties are assessed. Many plant basedecks may have more detail in the vessel component added to address specific needs for the AOO and stability applications. When available, these more-detailed base decks will be used to establish the baseline LOCA application for a particular plant. The methodology for validating the acceptability of changes to the baseline LOCA nodalization (once established for plant specific calculations) is discussed in Section 9.2 of the LTR.

As discussed in Section 9.2, [[

]] Once found to be acceptable, the results using the updated nodalization can be used for LOCA analyses.

[[

]] The updated nodalization strategy would then be validated similar to the validation of the generic nodalization (consistent with the TRACG Qualification Report (Reference R71-1). Once validated, the new nodalization can be used for LOCA analyses.

References

R71-1 GE Hitachi Nuclear Energy, "TRACG Qualification," NEDE-32177P, Revision 3, August 2007.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-72

The logic for selecting hot channels for TRACG simulations in the LTR prescribes [[

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However, particularly for BWRs with jet pumps, it is not clear that [[

]] Please justify that the existing hot bundle selection logic in the LTR is adequate to determine the limiting bundle for LOCA analysis or propose [[

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RAI-72 Response

Lower bundle MCPR increases the bundle power and higher LHGR increases the nodal power. Placing the bundle at these two target thermal limits assures that the PCT analysis includes the highest bundle and nodal powers.

The current TRACG LOCA experience shows that a higher bundle power for the top-peaked hot bundle is in general more limiting than a lower bundle power (higher axial peaking) if both bundles have the same LHGR at the peak power node. [[

]] Further implementation details are provided in the response to RAI-73.

As described in the RAI-73 response, the [[

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This change in the process will be applied to all plant types. Please also see the RAI-73 response.

LTR Impact

Section 6.2-5 of the LTR, including Table 6.2-2, is modified in the response to RAI-73 to include the requirement for the alternative top peaked bundle.

SNPB RAI-73

Please provide technical basis to support [[

]], addressing the following specifics:

a. Please summarize the source of the data (e.g., size of database, number of plants, approximate time period represented, fuel types, etc.).

b. Please clarify the expected differences in [[ ]] Please address in particular why [[ ]]

c. Please clarify what verification would occur during the core design process and/or operating cycle to ensure that [[ ]] is applicable to a given operating cycle for a particular plant and, [[ ]]

d. Regarding the limited data [[ ]], please justify that the scarcity of limiting bundles is supported by sufficient measurements in these regimes.

e. Given the [[ ]]

f. Please clarify the acceptable tolerance limit in the footnote to Table 6.2-2

RAI-73 Response

The responses are provided below to each part of this RAI. A revision is also made to the TRACG LOCA hot bundle selection criteria and the thermal limits applied to the hot bundles. Table R73-1 provided at the end of the responses replaces TRACG LTR Table 6.2-2.

a. The data in Figures 6.2-2 and 6.2-3 of the LTR include [[

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b. The process for selecting the hot bundle previously described in the LTR was [[ ]] The revised process is summarized in Table R73-1 which replaces TRACG LTR Table 6.2-2.

In general, a higher bundle power results in a higher PCT and/or oxidation given the same peak power node elevation and LHGR limit. [[

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The revised process is summarized in Table R73-1.

c. The bundle operating parameters that may affect the PCT or oxidation calculations are,

- [[

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f. Table 6.2-2 of the LTR is modified in Table R73-1 below. The table note that is the subject of this question no longer exists.

References

R73-1 Letter, J. S. Charnley (GE) to M. W. Hodges (NRC), "Proposed Amendment 19 to GE LTR NEDE–24011–P–A (Power Distribution Limits)," MFN 031-87, April 7, 1987.

LTR Impact

LTR Sections 6.2.5 and 6.2.8 are updated (Attached at end of this RAI response).

Table R73-1. Modified Table 6.2-2 from the TRACG LOCA LTR

[[				

]]

[[

]]

Figure R73-1 MAPRAT and Peak Power Node Number of Limiting Bottom Peaked Bundles

[[

]]

Figure R73-2 CPRRAT and Peak Power Node Number of Limiting Bottom Peaked Bundles

[[

]]

Figure R73-3 MAPRAT and Peak Power Node Number of Limiting Top Peaked Bundles

[[

]]

Figure R73-4 CPRRAT and Peak Power Node Number of Limiting Top Peaked Bundles

LTR markups for LTR Section 6.2.5 and 6.2.8 due to the response to RAI-73 are attached below:

In addition, LPF, defined as Local Pin Power Peaking Factor, should be added to the "ACRONYMS AND ABBREVIATIONS" table in the LTR.

6.2.5 Limiting Bundle Power Distribution

The limiting bundle LHGR and MCPR is of critical importance to the prediction of ECCS performance analysis critical safety parameters. The LHGR affects the stored energy in the fuel rod. The MCPR affects the proximity to boiling transition that is important in the early phase of many LOCA scenarios. The MCPR also influences the maximum bundle power because the MCPR limit constrains bundle power.

[[

]]

The term LOCA-limited in the following sections refers to plants with PCT values close to 1,478 K (2,200°F). These plants experience extended uncovery and rely on spray entering into the top of the bundle. In this sense, BWR/2s are LOCA-limited. LOCA-limited plants are highly sensitive to thermal radiation effects; therefore position of the fuel rods with higher peaking factors within the bundle is very important. [[

]]

Most jet-pump BWRs are not constrained by LOCA-related thermal limits. Non-LOCA limited plants are typically dependent on uncovery and recovery timing. PCTs are lower and results are less sensitive to radiation effects.

In the following sections, the process and basis for establishing target initial conditions for MCPR and LHGR (or MAPLHGR) is described. [[ ]]

Three limits constrain the design and operation of fuel bundles: Thermal Mechanical Operating Limit (TMOL), which is the limiting Peak Linear Heat Generation Rate (PLHGR), MAPLHGR and Operating Limit Minimum Critical Power Ratio (OLMCPR). Generally, it is not likely for a bundle to be at or near both the LHGR and MCPR limits. The limiting bundles are close to their maximum heat generation limits in early to mid-fuel cycle, when their axial power distribution is peaked near the bottom of the bundle. The limiting bundles are closest to the MCPR limit from the middle to the end of the fuel cycle, when their axial power distribution is mid or top-peaked.

[[

]]

The limits basis is reviewed for each cycle of application and may be updated based on the process defined above.

6.2.8 PLHGR, MCPR and MAPLHGR Uncertainty

As described in Section 6.2.5, initial conditions for LOCA evaluations conservatively assume that the fuel bundle in question is operating at the LHGR or critical power (MCPR) limit.

The 3D MONICORE core process computer takes total power, flow, pressure, and nuclear instrument signals from the reactor core and evaluates a peak LHGR, a peak average planar linear heat generation rate (APLHGR) for each six-inch node, and a CPR for each bundle in the core. There are uncertainties associated with each physical input to the process computer as well as the model used to evaluate the LHGR, APLHGR and CPR. [[

]] The monitoring uncertainties are included in the MCPR limits in accordance with NEDC-32694P-A [32], and in the Thermal Mechanical Operating Limit (TMOL) LHGR limit in accordance with NEDC-33258P-A [76].

[[

]]

Table 6.2-2 Summary of LHGR and MCPR Targets

[[				

]]

SNPB RAI-74

Please define [[

]]. Please particularly address why [[

]] Please further clarify whether [[

]], or some other reference value, and additionally justify that a consistent reference is used relative to [[ ]].

RAI-74 Response

[[

]] Additional discussion can be found in the responses to RAI-72 and RAI-73. The following process is followed for the bundles that are being set to the MCPR target.

[[

]]

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-75

Please clarify whether scram times under realistic LOCA conditions may be affected by the interference of control blades with core structures due to seismic- or LOCA-induced motion, or due to operational effects such as shadow corrosion-induced channel bow. If so, please clarify why an appropriate delay due to these effects need not be included in a best-estimate analysis.

RAI-75 Response

The scram time used in the analysis is the Technical Specification (TS) allowable value which bounds the variations that may occur during normal operation while the plant is operating in accordance with the licensing requirements. The Limiting Condition for Operation (LCO) and the surveillance requirements for scram time in the TS (LCO 3.1.4 in the Standard Technical Specifications) assure that any deviations from the nominal geometry during normal operation do not interfere with the scram time requirements. The control rod motion is tested during each power ascension from shutdown and at specified intervals during normal operation thereafter in accordance with the surveillance requirements in the TS.

The fuel design accounts for the seismic and LOCA induced motion. The third fuel acceptance criterion for LOCA in the Standard Review Plan (NUREG-0800) Section 4.2 Appendix A requires that "*Loads from the worst-case LOCA that requires control rod insertion must be combined with the SSE loads, and control rod insertability must be demonstrated for that combined load*." Compliance with this criterion is demonstrated in GESTAR II for GE and GNF fuels.

Therefore, no additional delay would need to be included in the scram time used in the analysis.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-76

[Follow-on RAI-12] A passage in the response to RAI-12 discusses the difference between offsite power assumptions employed in the generic demonstration cases as compared to those that may be used in an actual application. Among the LTR demonstration calculations, the information contained in the RAI 12 response, and the information displayed in LTR Table 2.5-1 (bottom of Page 2-6), the LTR appears to lack a succinct description explaining GEH's proposed treatment of offsite power availability in plant-specific applications. Please include a brief passage in the LTR that describes how plant-specific applications will ensure that the offsite/onsite power availability requirements of GDC-35 are addressed.

RAI-76 Response

As required by General Design Criteria 35, for emergency core cooling systems, "*suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.*"

In the response to RAI-12, specifically in Table R12-1, system availability was presented for the offsite power available assumption (OPA) and loss-of-offsite power (LOOP) assumption. The conservative assumptions as shown in Table R12-1 were used for LTR demonstration calculations. For future plant applications, [[
]] will be used to show the compliance to the General Design Criteria 35 requirement regarding onsite and/or offsite power.

The GEH process regarding "Single failure and loss of onsite and offsite power" in LTR Section 2.5.1 (bottom of Page 2-6) is modified.

LTR Impact

The GEH process regarding "Single failure and loss of onsite and offsite power" in Regulatory Position 3, Section 3.1 of LTR Table 2.5-1 is modified as shown below.

Original

Single failure and loss of onsite and offsite power should be considered.	Loss of preferred power is assumed. Sensitivities to single failures are considered.	Process conforms to Regulatory Guide and Appendix A of 10 CFR 50.

Revised

Single failure and loss of onsite and offsite power should be considered.	Consistent with the requirement of General Design Criteria 35, both loss of onsite power and loss of offsite power are assumed individually. System availability and system responses to loss of either onsite or offsite power is modeled [[ ]] ~~Loss of preferred power is assumed.~~ Sensitivities to single failures are considered.	Process conforms to Regulatory Guide and Appendix A of 10 CFR 50.

SNPB RAI-77

Please discuss the steady-state initialization process and what parameters and criteria are used to determine that the steady-state calculation has adequately converged prior to performing transient calculations.

RAI-77 Response

The steady-state characterizes an idealized condition that is assumed to be most representative of the initial state of the reactor system for analysis purposes. In reality, small fluctuations in all physical and monitored system parameters are always present. Furthermore, experience in performing LOCA calculations using thermal-hydraulic system codes shows that the effects of reasonably small departures from the intended steady-state conditions on the computed results are insignificant and, to a large extent, akin to the perturbations introduced by small variations in uncertainty parameters. Therefore, they are typically bounded by the uncertainty in the analysis. Nevertheless, a converged steady-state at the onset of the transient calculations is part of the process of performing computations using thermal-hydraulic codes, at least for the purposes of repeatability.

TRACG steady state calculations are performed [[

]]

Given the above features, it is sufficient to check a limited set of heat balance parameters to assure that the convergence is achieved. These parameters checked and their tolerances are listed below.

- [[

]]

Because the tolerance in initial steady-state can be treated as an independent source of uncertainty in the initial conditions, the effect on the total uncertainty in the initial conditions can be estimated [[

]]

If a deviation becomes necessary from the tolerances specified above, the deviation is justified on a case-by-case basis and demonstrated to have no effect on the results in the non-conservative direction.

Steady state convergence to the specified targets is assured by [[

]]

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-78

How does TRACG-LOCA account for potential uncertainties in the flow regime? Explain the analytic treatment for uncertainties associated with the transitioning from one type of flow to another?

RAI-78 Response

[[ ]] TRACG models of flow regimes and their transition from one regime to another are discussed in Section 5 of the TRACG Model Description LTR (Reference R78-1). [[ ]] It is important to note that the flow regime per se is not used by the TRACG field equations, but rather the values for the interfacial parameters. The main assessment of the flow regime map, therefore, should be done in connection with the interfacial shear model and based on the accuracy of the void fraction prediction. The flow regime potential uncertainties can then be represented in TRACG by how well TRACG predicts void fraction for BWR application and the uncertainties in flow regime predictions.

Interfacial shear modeling at different flow regimes are discussed in Section 6.1 of Reference R78-1. TRACG predictions of void fractions in varieties of test facilities can be found in the TRACG Qualification LTR (Reference R78-2). It has been shown in Reference R78-1 and Reference R78-2 that the void fraction is predicted by TRACG very accurately, [[ ]]

The void fraction potential uncertainties have been considered in the TRACG LOCA application by including those PIRT parameters that affect void fraction predictions, as described in LTR Section 3 and 5.

[[ ]]

References

R78-1 GE Hitachi Nuclear Energy, "TRACG Model Description," NEDE-32176P, Revision 4, January 2008.

R78-2 GE Hitachi Nuclear Energy, "TRACG Qualification," NEDE-32177P, Revision 3, August 2007.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-79

Section 5.3.2 of the LTR states that feedwater flashing is a dominant phenomenon in the latter part of the blowdown phase of the integral LOCA tests, yet the phenomena identification and ranking table (PIRT) treatment of the feedwater system appears to assign a medium importance rank. Both the FIST and ROSA facilities included feedwater piping. Please provide additional detail concerning the TRACG modeling of these facilities, and the treatment of the feedwater system within the TRACG models. Provide additional discussion specifically characterizing the observed impacts of feedwater flashing, and discuss the results of the TRACG assessments with respect to this phenomenon. Explain what conclusions are drawn with regard to the validity of the TRACG-LOCA EM and its treatment of jet pump BWR feedwater piping.

RAI-79 Response

The PIRT (LTR Table 3.4-1) was developed early in the Code Scaling Applicability and Uncertainty (CSAU) process for the TRACG LOCA application. At the conclusion of CSAU Step 3, the highest importance rank assigned to Item R1 (feedwater flow dynamics) was medium with [[ ]] as the critical parameter. It has been subsequently demonstrated [[ ]] (see the response to RAI-93 for further information [[ ]] Note that the biases and uncertainties associated with all high and medium ranked phenomena are considered in LTR Section 5.1. Thus, the treatment of this item is the same as if it had originally been ranked as high importance.

Section 5.3.2 of the LTR indicates that feedwater flashing is one of the important phenomena to consider, for purposes of test facility scaling, for simulating large break LOCA blowdown transients. The conclusion of this section is that because the cited tests generally scale well to the postulated BWR/6 large break LOCA, conclusions regarding the ability of TRACG to model these tests translates to the ability of TRACG to model the corresponding plant LOCA. [[ ]]

It is true that both the FIST and ROSA facilities included feedwater piping. However, note that in Section 5.3.2, feedwater flashing is only mentioned for the ROSA facility. For the FIST facility, the hot feedwater control/isolation valve is just upstream of the feedwater mixer/vessel inlet nozzle. Consequently, there is a very small volume of feedwater piping modeled outside the vessel ([[ ]]). Thus, feedwater flashing is under-scaled for the FIST facility relative to the corresponding plant case.

The ROSA facility Test 926 TRACG model includes the [[ ]] feedwater piping [[ ]] Because the vessel connection elevation is the high point of this pipe, feedwater delivery to the vessel will effectively cease once the feedwater is isolated, until feedwater flashing begins. [[ ]] The effect of feedwater flashing in this pipe is most readily observed by running the TRACG transient with and without the pipe, and comparing the system pressure response. Figure R79-1 provides this comparison, [[ ]]

[[ ]]

Figure R79-1 Impact of Feedwater Flashing on ROSA Test 926 TRACG Calculated Vessel Pressure

For a full-scale plant analysis with TRACG, it has been observed that feedwater flashing [[ ]] Figure R79-2 is included to compare the TRACG calculated peak rod temperatures for the same three cases previously discussed. This figure is included for information [[ ]]

As mentioned above, the FIST experiment did not include the feedwater flashing effect in any significant way. The feedwater flashing effect was included in the ROSA Test 926 experiment, but it was under-scaled relative to the corresponding plant case. However, the effect of this phenomenon on pressure was observed in the TRACG simulation, as in the experiment measured pressure. Despite this under-scaling of feedwater flashing and over-scaling of the metal structure stored energy in the experiments, as LTR Section 5.3.2 indicates, the overall scaling parameters are reasonable for the corresponding plant LOCA analysis with TRACG. Finally, the ability of TRACG to calculate liquid flashing during depressurization has been shown by separate effects tests such as the PSTF level swell test (Section 3.1.5 in Reference R79-1) and various critical flow tests (Section 3.4 in Reference R79-1).

[[

]]

Figure R79-2 Effect of Feedwater Flashing on ROSA Test 926 TRACG Peak Rod Temperature

References

R79-1 GE Hitachi Nuclear Energy, "TRACG Qualification," NEDE-32177P, Revision 3, August 2007.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-80

Please clarify the Steam Sector Test Facility test used in Figure 5.3-3.

RAI-80 Response

The test data for SSTF in LTR Figure 5.3-3 are from SSTF Test SRT-3 Run 26 for the bundle with counter-current flow, which can be found in Figure 3.4.4-22 (Reference R80-1). [[

]]

SSTF Test SRT-3 (Run 26) simulated the reflood phase of a large-break LOCA transient. The detailed discussion of this test and its comparisons with TRAC and TRACG can be found in References R80-1 and R80-2.

References

R80-1 "BWR Refill-Reflood Program Task 4.8 – TRAC-BWR Model Qualification for BWR Safety Analysis Final Report," GEAP-22049 (NUREG/CR-2571; EPRI NP-2377), July 1983.

R80-2 GE Hitachi Nuclear Energy, "TRACG Qualification," NEDE-32177P, Revision 3, August 2007.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-81

Please explain the difference between the early boiling transition peak calculated in TRACG in Figure 5.3-8 for the ROSA test and the measurements that do not show a peak.

RAI-81 Response

The ROSA measurement shown in LTR Figure 5.3-8 is the ROSA data with 2nd peak from ROSA Test 926. The same data can be found in Figure 5.4-9 of Reference R81-1.

The early boiling transition PCT peak (called 1st peak) from ROSA test 926, as shown in Reference R81-1, is not included in LTR Figure 5.3-8 because the maximum PCT is the most important parameter for LOCA application. [[

]]

Reference

R81-1 GE Hitachi Nuclear Energy, "TRACG Qualification," NEDE-32177P, Revision 3, August 2007.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-82

Section 5.3.3 of the LTR discusses the scaled integral LOCA simulation tests for non-jet pump plants. This discussion is supplemented by Section 5.5 of NEDC-32177P, Revision 3. In particular, Section 5.5 of NEDE-32177P notes that neither of the two integral tests described therein involved ECCS actuation. It is not clear that the regulatory guidance is satisfied for this reactor design. In particular, Standard Review Plant (SRP) 15.0.2 notes that "Integral effects testing must be performed to demonstrate that the interactions between different physical phenomena and reactor coolant system components and subsystems are identified and predicted correctly." As the LTR refers to a suite of integral effects tests as supplemented by additional separate effects tests, explain how the TRACG-LOCA EM is qualified in an integral sense. One important aspect, for example, is the behavior of non-condensibles in the vessel and primary system.

RAI-82 Response

The separate effects tests (SETs) and integral effects tests (IETs) play an important role in nuclear reactor thermal-hydraulics research. They are relied on developing the methodologies used in safety analyses. As stipulated in NUREG-0800 (Reference R82-1), the Standard Review Plan, and also exemplified in NUREG/CR-5249 (Reference R82-2), the CSAU Methodology, and subsequently in Regulatory Guide (RG) 1.157 (Reference R82-3), the IETs primary focus is on the interaction between parameters and processes by incorporating many or all of the important phenomena and components. In the art of computational thermal-hydraulics, there has been always a residual imperfection when it comes to integral tests. In some cases, this is caused by the scaling distortion, in some others mainly based on facility restrictions. Compared to an actual plant data, IETs cannot be considered 'perfect'. Furthermore, it is not unusual to experimentally simulate portions of a LOCA event, rather than the entire transient in the integral effects testing. The referenced IET in Section 5.5 of Reference R82-6 is such an example that includes only the blowdown portion of the LOCA for a scaled external loop BWR prior to ECCS activation. During the blowdown the interactions between phenomena are complicated and that is why IETs are important during this stage. After blowdown the ECCS phenomena for a BWR/2 ECCS are limited to those represented by the core spray heat transfer SETs described in Section 3.2.2 of Reference R82-6. Although the entire system is not simulated, the key interactions related to spray distribution and heat transfer mechanisms within the fuel channel are captured. This is an example of the common practice of using SETs to address any gaps in the IET coverage.

One important distinction in use of IETs between the RG 1.157 guidance and the TRACG LOCA methodology is in how they contribute to quantification of code uncertainty. RG 1.157 states that the "*code uncertainty should be evaluated through direct data comparison with relevant integral systems and separate-effects experiments at different scales*". The approach employed in the TRACG LOCA methodology primarily uses the SETs (rather than the IETs) to quantify the biases and uncertainties in specific modeling phenomenon. The primary role of IETs in the

TRACG LOCA methodology is to evaluate an overall quality of the predictions considering the interaction between the modeled phenomena. Similar points are also provided as part of the RAI-92 response.

In this context, it is also prudent to recognize that, in some cases, not all of the important phenomena were identified and factored in, a priori, when the integral tests were originally designed. The existence of some interactions or the effect of some phenomena that were not recognized as important during testing can be identified by the realistic simulations using capable prediction tools. A recent example in light water reactor (LWR) research and best-estimate LOCA field is the discovery of the importance of downcomer boiling in pressurized water reactors (PWRs). As indicated in Reference R82-4, downcomer boiling was not recognized as a process of high importance in the CSAU study, an aspect that was established before the experimental information from the two-dimensional (2D)/ three-dimensional (3D) test programs was available and well before realistic code simulations were performed and ultimately, causing experimental data from large scale test facilities ending up being limited.

Similarly, in the case of BWR/2 analyses, the importance of air ingress and behavior of non-condensable gas (NCG) in the vessel and the primary system is revealed by TRACG LOCA calculations rather than any particular IET (as documented in the response to RAI-13). Because the importance of such behavior was not highlighted when the IETs were designed, no particular accommodations were considered in the IETs. Therefore, the experimental data in the sense of integral testing is limited. To this extend, the observation offered in the RAI has merits. The TRACG LOCA methodology, similar to other state-of-the-art LOCA methodologies, relies on complementing information gaps that might exist from the IETs and plant data by SETs and component data. The identified gap of no NCGs in the IETs is not a critical issue for the methodology, because the non-condensable effects on physical phenomena such as condensation are well established by the TRACG LOCA methodology. Section 5.1.5.3 of the TRACG LOCA LTR describes how the uncertainty in the degradation in condensation due to the presence of NCGs was established from SETs. Section 6.6.11 of Reference R82-7 describes the model in detail and comparison of the model to other models and the data is documented in Section 6.6.11.3. As shown by this example, the quantification of bias and uncertainties on the key phenomenon of degradation of condensation heat transfer due to NCGs does not directly rely on IETs because this has been achieved using SET data instead.

The core spray test comparisons described in Section 7.4 of the TRACG LOCA LTR include the phenomenological interactions relevant for the ECCS portion of a LOCA in a non-jetpump BWR. Selected comparison of calculated and measured rod temperatures shown in the LOCA LTR demonstrate that the uncertainties considered are sufficient. Additional core spray results were provided in the response to RAI-92. Even more additional experimental data are evaluated here as part of this response. The additional data supplied in this response comes from the Studsvik core spray heat transfer tests (Reference R82-5). The facility could be configured in different ways. As shown in Figure R82-1, the setup for the core spray heat transfer tests allowed bottom venting. The vents were open to atmosphere. Although the tests were not aimed to study air effects, during the venting, it was possible for air to enter the test rig unrestricted.

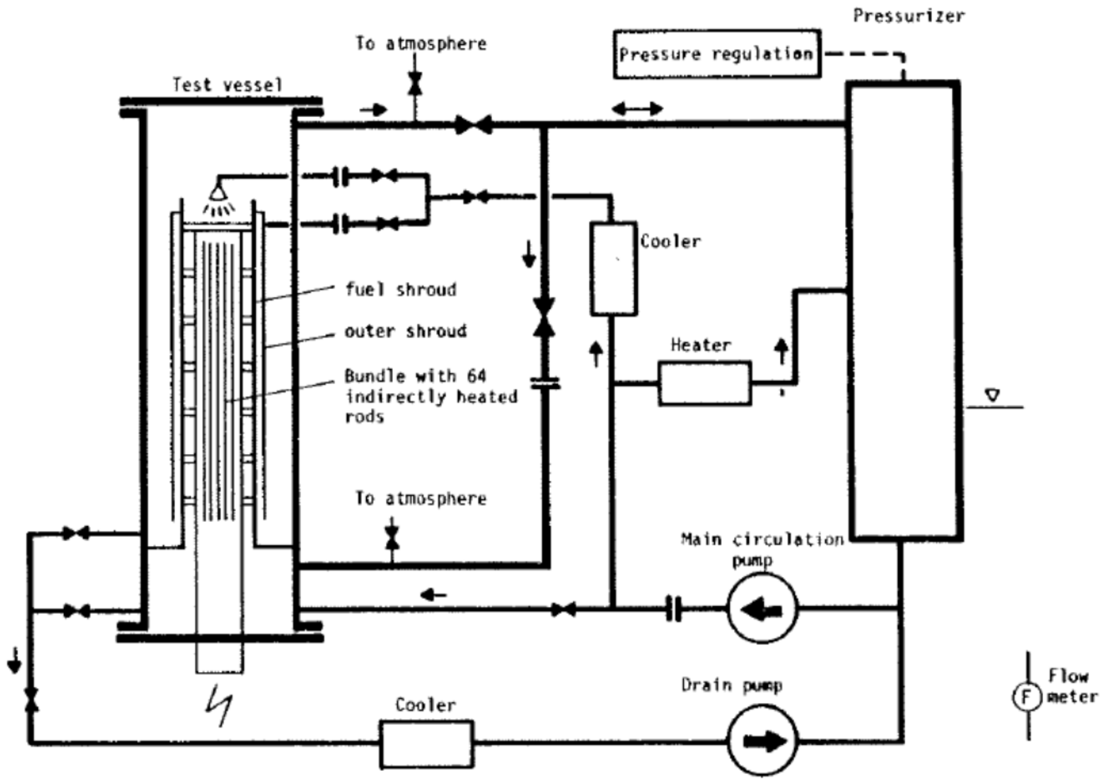

Figure R82-1 Spray Cooling Experiment Setup

A series of tests were conducted top and bottom venting configurations. The test run 123 was specifically designed as bottom vented. No measurements were made to determine how much air was ingested; however, the TRACG calculations for this test indicate [[ ]] as shown in Figure R82-2. Note that the calculational process to simulate conditions with and without possible air ingestion was achieved [[ ]] the same as used in the BWR plant simulations. The response to RAI-13 explained in detail the modeling approach used to ensure that effects of air ingestion would be reflected in the calculated PCT values used for the licensing basis.

[[

]]

Figure R82-2 Studsvik CSHT Test 123 – Cladding Temperatures

In this run, [[

]]

Additional core spray heat transfer tests were also evaluated by statistical treatment of uncertainties, sampling them from their respective distributions. Although these tests were designed as top venting, bottom venting was allowed in a controlled manner as the lower plenum was filled during the experiments. Three tests were selected because they had the highest peak temperatures. Figures R82-3, R82-4, and R82-5 show the comparisons for tests 111, 133, and 137, respectively.

[[

]]

Figure R82-3 Studsvik CSHT Test 111 – Cladding Temperatures

[[

]]

Figure R82-4 Studsvik CSHT Test 133 – Cladding Temperatures

[[

]]

Figure R82-5 Studsvik CSHT Test 137 – Cladding Temperatures

Combined with the statistical treatment, the TRACG results are [[

]] The conclusion is that the qualification basis is sufficient to support how the TRACG LOCA methodology is applied for non-jet pump plants.

References

R82-1 U.S. NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800, March 2007.

R82-2 B. Boyack, et al., "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss of Coolant Accident," NUREG/CR-5249, December 1989.

R82-3 U.S. NRC, "Best-Estimate Calculations of Emergency Core Cooling System Performance," Regulatory Guide 1.157, May 1989.

R82-4 NRC Memo S. Bajorek to J. E. Rosenthal, "Downcomer Boiling Technical Summary", May 22, 2002 [ML021420196], and its attachment "Downcomer Boiling" [ML021420201].

R82-5 Studsvik Report, "BWR Emergency Core Cooling Investigations – Spray Cooling Heat Transfer Experiment in a Full Scale BWR Bundle Mock-up," STUDSVIK/E4-78/64, October 4, 1978.

R82-6 GE Hitachi Nuclear Energy, "TRACG Qualification," NEDE-32177P, Revision 3, August 2007.

R82-7 GE Hitachi Nuclear Energy, "TRACG Model Description," NEDE-32176P, Revision 4, January 2008.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-83

Please clarify whether sensitivities associated with fuel channel grouping were performed for the LOCA event as indicated in Section 6.1 of the LTR and summarize the analysis and results, particularly as pertaining to [[
]].

RAI-83 Response

The changes in [[

]] Results, summarized in Table R83-1, show that the effect on computed PCT is negligible with the average channel groups.

Table R83-1 PCT Comparison for Channel Grouping Sensitivity

	[[	**]]**
BWR/4		
Double-Ended Guillotine Break	[[	
Intermediate Break (0.67 ft^2)		
Small Break (0.10 ft^2)		]]
BWR/2		
Discharge DBA	[[	]]

Based on the runs from this time step sensitivity study, the comparison of PCT results between the [[

]]

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-84

Please clarify whether analysis is required for the increased core flow region of the power/flow map. If not, explain why not.

RAI-84 Response

Consistent with the response to RAI-27, the increased core flow region will be evaluated to determine if the analysis is required for a specific application.

In general, [[

]]

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-85

Some LOCA-limited plants may not be BWR/2s; please ensure Table 6.2-2 reflects this consideration.

RAI-85 Response

[[
]]

Please note that Table 6.2-2 is updated in response to RAI-73.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-86

Section 6.3 of the LTR indicates that more realistic distributions may be used for plant parameters than specified in Tables 6.3-1 and 6.3-2 if justified separately for plant-specific analysis. Please provide the following additional information regarding this topic:

a. Please clarify whether all parameters in Table 6.3-1 and Table 6.3-2 may be substituted with more realistic distributions, or only a subset thereof, and justify that the data supporting more realistic distributions taken under normal conditions is relevant to performance during a LOCA (e.g., scram times, pump coastdown times, etc.).

b. Please clarify the statistical requirements to support the use of more realistic distributions for plant parameters.

RAI-86 Response

(a) A number of listed parameters in LTR Table 6.3-1 are physical parameters that define the plant configuration and are not changeable. Among these are total number of valves and grouping of these valves. As the table indicates, they are modeled based on the actual plant configuration. In the context of system performance and setpoint values, these parameters do not need to be listed and will be removed from the table.

Table 6.3-1 as modified by this response primarily relies on use of analytic limits (ALs) because the use of the established AL removes the need to establish and sample from a distribution. Using the established ALs ensures consistency between different functions when a trip is shared. Those parameters that currently use a basis other than the AL have already been justified in the LTR.

Consistent with the realistic modeling assumptions, other parameters given in the table that are currently set at the AL may have the AL replaced with a more realistic distributions based on the plant's approved instrument setpoint methodology. If the plant in question has not adopted an improved setpoint methodology program, then the justification of data for supporting more realistic distribution will be provided for approval.

(b) The NRC-approved instrument setpoint methodology program provides the basis for developing realistic distributions for values of plant parameters (Reference R86-1). The nominal parameter values and distribution functions would cover at least 95% content with at least 95% confidence. A different method for generating more realistic distribution for LOCA use for any specific parameter currently using an AL will constitute a change in the application model and thus require that the change for that specific parameter to be submitted for approval.

Reference

R86-1 GE Nuclear Energy, "General Electric Instrument Setpoint Methodology," NEDC-31336P-A, Revision 0, September 1996.

LTR Impact

Table 6.3-1 will be updated as marked below:

System Unique Parameters	Analysis Basis
Maximum delay time from diesel generator (DG) start signal until bus is at rated voltage	[[
Minimum detectable break size for Loop Selection Logic	
Leakage allowance for shroud access hole cover repairs or cracks	
Scram speed to 90% position	
Feedwater pump coastdown (from initial value to zero flow)	
Time constant for recirculation pump coastdown (intact loop and loop with break)	
ECCS makeup water temperature	
ADS Parameters Timer delay Bypass timer delay for sustained low water level ~~Total number of relief valves with ADS function~~ ADS close on vessel pressure ADS reopen on vessel pressure ADS reclose on vessel pressure	
Pilot-Actuated Safety/Relief Valves (SRVs) ~~Number in each Setpoint Group~~ Setpoints ~~Low-low set logic assumed in analysis for closing/opening pressure~~ Pilot-actuated SRV Capacity at (100+ACC)% of Popping Pressure Closing pressure setpoint Time delay before opening Time constant of opening/closing	
Spring Safety Valves (SSVs) ~~Number of groups~~ ~~Number of SSVs~~ Opening setpoint Capacity of each at opening setpoint	]]

SNPB RAI-87

Please clarify the code and LTR methodology (if applicable) used in the prediction of the containment pressure values used to characterize the drywell high pressure scram time, and contrast the results to times derived from existing licensing basis calculations. Please further clarify whether the hypothesis of normality was invoked in determining the 95/95 tolerance limit based on seven sample calculations performed at each break size.

RAI-87 Response

Drywell pressurization analysis was performed with the GE methodology used in sizing the vacuum breakers (VACBR code). [[

]] The results are compared in the table below. Because the initial containment pressure assumptions are also different in the licensing analyses, the comparisons are presented in terms of the time required to increase the drywell pressure by the same differential pressure. [[

]]

This simplified approach allows us to credit the delayed scram in the LOCA analysis in a conservative way without the complexity of coupled modeling of the containment.

Break Size (ft^2)	Time to Increase DW Pressure by 2 psid			
	LTR Figure 6.3.1	Containment Licensing Method Results	Containment Licensing Method	LTR Correlation
[[				
				]]

[[

]]

LTR Impact

No changes to the LTR are made as the result of this RAI response.

SNPB RAI-88

Please clarify whether the entries in Table 6.3-1 for automatic depressurization system (ADS) close/reopen/reclose on vessel pressure refer to the relief valve mode of operation of the safety relief valves (SRVs) used by the ADS. If not, please explain the intent.

RAI-88 Response

The ADS close/reopen/reclose on vessel pressure entries in LTR Table 6.3-1 refer to the pressure at which the SRVs operated by the ADS in relief mode will either: close when the differential pressure between the vessel and the containment is not sufficient to keep the valves open (after depressurization) or open when the differential pressure becomes high enough to reopen the valves.

LTR Impact

No changes to the LTR are made as the result of this RAI response.

SNPB RAI-89

[Follow-on to RAI-31] In Section 7.3, please clarify why only the limiting nominal conditions (i.e., without consideration of uncertainty) are evaluated statistically. Because the uncertainty may vary substantially among different breaks, the limiting conditions with respect to the 95/95 tolerance limits used to assess compliance with the criteria of 10 CFR 50.46 may not necessarily correspond with the conditions that are nominally limiting. Further, the process used to obtain the biased results discussed in response to Set 1 RAI-31 is not clear. Finally, it would seem that a better way to address the concern would be to re-evaluate the demonstration analyses for several break sizes in close proximity to the nominally limiting break sizes to demonstrate the insignificance of identifying the limiting break size (and other properties) using a nominal analysis. Presently, LTR Figure 8.1-29 underscores the concern conveyed in this RAI question. Please base the justification provided in response to this RAI question on updated break spectrum analyses and explain whether the more detailed channel grouping has improved the TRACG-LOCA EM performance in this regard.

RAI-89 Response

The GEH approach is consistent with the requirements of Section 4.4 of RG 1.157, which requires that "*the evaluation of the peak cladding temperature at the 95% probability level need only be performed for the worst-case break identified by the break spectrum analysis in order to demonstrate conformance with paragraph 50.46(b)*".

First, we discuss the process used to obtain the biased results in the response to Set 1 RAI-31. The biases referred in that response were obtained from the comparisons between TRACG predictions and test data, which were determined individually for each PIRT parameter and were discussed in details in Section 5 of the LTR (See a summary in Table 5.1-2 of the LTR). The biased results in the response to Set 1 RAI-31 refer to the TRACG calculation results in which those TRACG model biases determined in Section 5 of the LTR are not removed. On the contrary, the non-biased results in the response to Set 1 RAI-31 refers to the results in which the TRACG model biases are removed. The non-biased results are actually, more precisely, de-biased results, which will be called hereafter.

It is acknowledged that [[

]] (See the discussion in the response to RAI-6).

Regarding the results in LTR Figure 8.1-29, [[ ]], as shown in the figure. LTR Section 7.3 provides the method for determining the One Sided Upper Tolerance Limit (OSUTL). Because it is the OSUTLs that are compared to the acceptance criteria, the reasonable and appropriate comparison between different breaks is for OSUTL values, not for maximum values. [[ ]]

For a BWR/4, the PCT variability has been significantly reduced [[ ]] Sensitivity studies are performed here to demonstrate that the PCT determined from the limiting break determined from the nominal calculation is sufficient for TRACG application. For this study, a BWR/4 is selected. As shown in the response to Set 1 RAI-31, [[ ]] The results from all 10 sets of the analyses are summarized in Table R89-1.

As shown in Table R89-1, [[ ]] (See Figure R31-1 in the response to Set 1 RAI-31).

It is found that [[ ]] TRACG-LOCA EM performance has been improved in predicting the OSUTL PCT in the close vicinity of the limiting break. This further demonstrates that the statistical analysis at the limiting break sizes determined from nominal conditions can sufficiently represent the results in the close proximity of the limiting break.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

Table R89-1 PCT Summary of Sensitivity Study for BWR/4 Suction Line Breaks

[[						
						]]

SNPB RAI-90

Section 7.3.1 states a minimum bound for the number of simulations that may be increased to raise the confidence level of the desired statistical bound. Please clarify whether GE Hitachi Nuclear Energy – Americas, LLC (GEH) will require that the number of simulations be set prior to performing analysis in order to prevent degradation of the statistical confidence level.

RAI-90 Response

The number of simulations that satisfy the 95^{th} percentile with 95% confidence is set prior to performing the analysis. [[

]]

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-91

Please clarify the statement in Section 7.4.4 that TRACG underpredicts mixing in the VSSL component. It appears that the spread in lower plenum temperature predictions in the TRACG results are less than the data, and that the mean value of the lower plenum temperature is underpredicted by TRACG.

RAI-91 Response

The benchmarking to the SSTF had been updated using the current TRACG04 version and the updated PIRT table. The discussion in LTR Section 7.4.4 and Figures 7.4-10 through 7.4-17 are revised to include the updated results.

The lower plenum temperature distribution has an indirect effect on the LOCA response because the temperature of the water going into the core is nearly at saturation temperature. Stratification in the lower plenum may affect the LOCA response through the pressure and the amount of mass contained in the lower plenum. The revised LTR includes comparisons to the water mass in the lower plenum, core bypass region, upper plenum, system pressure and the temperature distribution. The statement that is the subject of this RAI is replaced by a more extensive discussion.

LTR Impact

LTR Section 7.4.4 (including figures) is updated by this RAI response as indicated in the following markups.

7.4.4 SSTF Test EA3-1

The SSTF Test Facility simulated the refill-reflood phase of a BWR LOCA transient. The safety systems in the facility could be configured to model either a BWR/6 or BWR/4 refill-reflood transient and test data from both BWR/6 and BWR/4 simulations were used for TRACG qualification [2]. Test EA3-1, a BWR/4 refill-reflood simulation, was selected for the statistical analysis presented here.

In this test, the lower plenum fill time controls the core reflood. The major purpose of the statistical analysis is to show that the measured lower plenum fill-time is bounded by the TRACG analysis when model uncertainties are taken into account. The liquid mass held up in the bypass region and upper plenum has an important effect on the LOCA analysis. The bypass and upper plenum fill time predictions are compared to the test results. It should be noted that the accuracy of the lower plenum fill time also depends on the amount of liquid held up above the core plate, and therefore the lower plenum fill time predictions depend on the bypass and upper plenum liquid fractions being predicted reasonably well. ~~A secondary purpose is to examine the effect of model uncertainties on the prediction of mixing of the cold ECCS injection in the lower plenum during refill.~~ Several of the qualification test data vs. analysis comparisons are repeated as part of the uncertainty analysis.

The results of the statistical analysis of SSTF Test EA3-1 are compared with the test data in Figure 7.4-10 through Figure 7.4-~~14~~17. The TRACG results include the nominal calculation, the band subtended by the 59 trials and the average of the 59 trials. Figure 7.4-10 compares measured and calculated system pressures. [[

]]

TRACG's prediction of ~~the mixing of the cold ECC water injected into~~ temperature variations in the lower plenum is shown in Figure 7.4-11 through Figure 7.4-13. These figures show temperatures measured at the lower plenum periphery near the bottom, at mid-submergence and near the surface. The data for the bottom and mid-plane show the same trend until approximately 80 seconds, after which there are fluctuations of high magnitude at the mid-level and of lower magnitude at the bottom level. ~~At the lower plenum bottom, where the injection takes place, TRACG predicts [[~~

Since the lower section of the lower plenum is divided into bays, and the temperature is measured in the bay where the jet pump discharges into, the lower plenum temperature measured at the bottom of the lower plenum is expected to be close to the jet pump discharge temperature. There are occasional fluctuations in the test data after the lower plenum is filled as shown in

Figure 7.4-11. [[

]]

Of particular interest in Test EA3-1 is the lower plenum refill time. The available test data show the refill time expressed as lower plenum mass fraction. ~~The TRACG lower plenum mass fraction was calculated from the cell mixture density.~~ [[

]]

Comparisons to the bypass and upper plenum fill fractions in Figures 7.4-16 and 7.4-17 provide a measure of the effects of CCFL correlations. [[

]]

These comparisons show that the TRACG predictions represent the ECCS mixing phenomena reasonably well in the upper plenum and lower plenum regions, CCFL behavior and the timing of reflood.

The uncertainty evaluation of the TRACG predictions of the SSTF test was extended to consider the potential range of the test conditions along with the model parameters in the Monte Carlo calculation. Uncertainties in the test conditions include the flow rates and temperatures of the ECC systems, the temperatures and flow rates of the steam injections that simulate vapor generation and the initial liquid masses in the various regions of the test facility. [[

]] The test conditions that were varied together in this manner are noted in Table 7.4-1.

Figure 7.4-~~16~~18 and Figure 7.4-~~17~~19 show the predictions for the pool temperature at the periphery near the bottom and for the lower plenum mass fraction when both the model parameters and test conditions are varied. [[

]]

In summary, the statistical analysis of SSTF Test EA3-1, a BWR/4 refill-reflood simulation,
[[

]]

[[

]]

Figure 7.4-10 Comparison of System Pressure for SSTF Test EA3-1

[[

]]

Figure 7.4-11 Comparison of Temperature at Pool Periphery at the Bottom for SSTF Test EA3-1

[[

]]

Figure 7.4-12 Comparison of Temperature at Pool Periphery at Mid-Depth for SSTF Test EA3-1

[[

]]

Figure 7.4-13 Comparison of Temperature at Pool Periphery at the Surface for SSTF Test EA3-1

[[

]]

Figure 7.4-14 Comparison of Lower Plenum Fill Fraction for SSTF Test EA3-1

[[

]]

Figure 7.4-15 Correlation of Lower Plenum Refill Time with PIRTs for SSTF Test EA3-1

[[

]]

Figure 7.4-16 Comparison of Bypass Fill Fraction for SSTF Test EA3-1

[[

]]

Figure 7.4-17 Comparison of Upper Plenum Fill Fraction for SSTF Test EA3-1

[[

]]

Figure 7.4-18 Comparison of Temperature at Bottom of Pool Periphery Including Variations in Test Conditions for SSTF Test EA3-1

[[

]]

Figure 7.4-19 Comparison of Lower Plenum Fill Fraction Including Variations in Test Conditions for SSTF Test EA3-1

SNPB RAI-92

In Section 7.4.5, Core Spray Heat Transfer (CSHT) Test 111 was chosen for comparison with TRACG results to justify the model parameter uncertainties and statistical combination process. Please address the following associated issues:

a. Although most data for CSHT Test 111 is bounded by TRACG predictions, the peak temperatures for the hottest rods significantly exceed the mean of the TRACG predictions and in some cases even exceed the maximum value of the 59 TRACG predictions (e.g., Figure 7.4-18). Since it is the peak values that are of regulatory interest, please justify the LTR's conclusion that uncertainties proposed for modeling core spray are adequate.

b. Please justify the selection of CSHT Test 111 for comparison and discuss whether the conclusions made in Section 7.4.5 of the LTR would hold if comparisons were instead made with additional CSHT tests such as Tests 112 and 121 (see Section 3.2.2 of the Qualification Report). Demonstrating that proposed uncertainty distributions provide a representative bound on tests such as CSHT is vital due to the lack of integral testing involving extended heatups characteristic of LOCA-limited BWRs.

RAI-92 Response

Before the justification and clarification is made for this RAI, it is worthwhile to emphasize the purpose of those statistical analyses presented in LTR Section 7.4. As discussed in the beginning of this section, "*in this section, statistical analyses are performed on a selected set of ECCS/LOCA qualification tests utilizing the model uncertainties developed in Section 5. Statistical analyses of the integral system tests validate the values used for the model uncertainties by showing that the test data fall within the resulting uncertainty band of the calculations.*" The statistical analyses in Section 7.4 are not used to justify the model parameters, which are determined individually using existing test data, as discussed in detail in LTR Section 5. To achieve this purpose, the same statistical process as that used for the Section 8 demonstration analyses is applied to those tests using the same model parameters and their uncertainties (determined in Section 5 of the LTR). The test-specific model parameter adjustments (biases) are not used.

It should be noted that the information presented in LTR Section 7.4 was from the draft LTR, and the updated information was not implemented in the submitted LTR. This has been captured in the GEH Condition Report process as CR 8235. The CSHT results presented in this RAI are updated with the current GEH Level 2 TRACG04P and updated PIRT table.

<u>Response to Part a:</u>

CSHT Test 111 was re-run with the current Level 2 TRACG04 and the updated TRACG LOCA PIRT table. The same process as described in Section 7.4.5 of the LTR is used. In addition to the TRACG model parameter uncertainties, the analysis also considers [[

]] The key outputs from the test are the temperature responses for various rods in the bundle at the peak power elevation.

Figure R92-1 through Figure R92-5 show the results of the analysis in terms of the temperature of five TRACG rod groups corresponding to the locations of the test measurements. [[

]] See further discussion in the Summary section of this RAI response.

Response to Part b:

There are total of six CSHT tests in Reference R92-1. As shown in Table 3.2-3 of Reference R92-1, Test 111 has the second highest initial rod PCT, and it has complete test data. In addition, [[ ]], as shown in Table 3.2-4 of Reference R92-1. Therefore Test 111was selected in the original LTR for testing both rod PCTs and rod quenching characteristics.

CSHT Test 112 has the highest initial rod PCT, among all six tests (Table 3.2-3 of Reference R32-1), but this test only has test data up to 500 seconds. The comparison of this test for quenching with TRACG is not possible, and it was therefore not selected in the LTR. Test 121 was not selected for the LTR due to the low initial rod PCTs. In addition, [[ ]] (see Table 3.2-4 of Reference R32-1).

In response to this RAI, it was determined that the addition of TRACG [[

]]

CSHT Test 112 was run using the same process as that for Test 111. Figure R92-6 through Figure R92-8 show the results of the analysis in terms of the temperature of four TRACG rod groups corresponding to the locations of the test measurements. [[

]] The situation for Test 112 is similar to Test 111 above.

Summary

TRACG predictions of the peak rod temperature for both Test 111 and Test 112 with the TRACG LOCA model uncertainties established in the LTR provide the representative bound for the rod temperatures from the tests. It is also observed that [[

]]

It is acknowledged that [[

]]

It is further noted that [[

]]

In summary, TRACG predictions of the peak rod temperatures for [[

]]

Reference

R92-1 GE Hitachi Nuclear Energy, "TRACG Qualification," NEDE-32177P, Revision 3, August 2007.

LTR Impact

LTR Section 7.4.5 will be updated.

[[

]]

Figure R92-1 Results of Monte Carlo Analysis vs. Test Data for Rod Group 2

[[

]]

Figure R92-2 Results of Monte Carlo Analysis vs. Test Data for Rod Group 3

[[

]]

Figure R92-3 Results of Monte Carlo Analysis vs. Test Data for Rod Group 4

[[

]]

Figure R92-4 Results of Monte Carlo Analysis vs. Test Data for Rod Group 6

[[

]]

Figure R92-5 Results of Monte Carlo Analysis vs. Test Data for Rod Group 9

[[

]]

Figure R92-6 Results of Monte Carlo Analysis vs. Test Data for Rod Group 2

[[

]]

Figure R92-7 Results of Monte Carlo Analysis vs. Test Data for Rod Group 3

[[

]]

Figure R92-8 Results of Monte Carlo Analysis vs. Test Data for Rod Group 4

[[

]]

Figure R92-9 Results of Monte Carlo Analysis vs. Test Data for Rod Group 6

[[

]]

Figure R92-10 Results of Monte Carlo Analysis vs. Test Data for Rod Group 2

[[

]]

Figure R92-11 Results of Monte Carlo Analysis vs. Test Data for Rod Group 3

[[

]]

Figure R92-12 Results of Monte Carlo Analysis vs. Test Data for Rod Group 4

[[

]]

Figure R92-13 Results of Monte Carlo Analysis vs. Test Data for Rod Group 6

SNPB RAI-93

Regarding the statement in Section 8.1.1 that [[
]], please clarify the following:

a. [[ ]]

b. Please provide the basis for terminating the model at this point (e.g., post-LOCA system isolation point, sensitivity calculations demonstrate no further impact from extending model, etc.).

c. Please justify that the modeling of the feedwater system is either best-estimate or conservative.

RAI-93 Response

a. The 'first' feedwater heater refers to [[
]]

b. This is a convenient termination point because [[

]] The results are summarized in the following table.

Table R93-1 Feedwater Additional Volume Sensitivity Results

[[						
						]]

[[

]]

c. Modeling of the feedwater piping [[

]]

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-94

[Follow-on to Set 1 RAI-20] The response to Set 1 RAI-20(a) states, in part, [[

]] Please clarify whether [[

]]

RAI-94 Response

Regarding the response to Set 1 RAI-20, it should be emphasized that the current core spray flow distribution presents an extreme or a limiting case that bounds any possible value. [[

]]

The expected insignificant effect on LOCA parameters has been demonstrated by sensitivity studies.

[[

]]

Reference

R94-1 General Electric, "Core Spray Design Methodology Confirmation Tests," NEDO-24712-A, March 1983.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

Table R94-1 PCT Summary of Sensitivity Study For BWR/4 Suction Line Breaks with Different Spray Distributions in TRACG Ring 1 to 3 (0.5 ft^2 Recirculation Suction line break)

[[						
						]]

SNPB RAI-95

Section 8.1.2.2 states that similar to results from the SSTF (e.g., as discussed in NUREG/CR-2566), it is possible for bundles in TRACG simulations to make a transition from the one mode [[ ]] to the other during the transient, and that the results confirm TRACG's capability to predict this behavior. Please clarify this statement relative to the results in Section 6.4 (particularly Figures 6.4-7 and 6.4-8), which appear to suggest very limited potential for transition between states beyond an initial bifurcation point.

RAI-95 Response

The discussion in the LTR Section 8.1.2.2 is for a small break. The sentence in this section that "*it is possible for bundles to make a transition from one mode to the other during the transient*" refers to the [[ ]] The last paragraph of this section discussed the PCT bifurcation due to the [[ ]] phenomenon, which is more related to the discussion in Section 6.4, and should not have been in Section 8.1.2.2. This paragraph will be revised in the LTR.

As described in NUREG-CR-2566 (Reference R95-1), the "parallel channel effects" refers to the observations that all fuel channels in the core are not necessarily at the same flow mode following a jet-pump BWR LOCA. It is further indicated in NUREG-CR-2566 that "parallel channel effects" can only occur when there is a two-phase level in the lower plenum, which allows re-distribution of the low plenum steam to the channel inlet orifices. As discussed in Reference R95-1, especially in the Section 5 discussion, which is pertinent to a jet pump BWR LOCA, "parallel channel effects" are evidenced by three different channel flow modes that may occur simultaneously in the SSTF tests (SRT 3 Run 26), which are: (1) Counter-current flow, (2) Co-current upward flow, and (3) Co-current down flow. The capability of TRACG modeling the "parallel channel effects" from SSTF was demonstrated in TRACG qualification LTR Section 5.3.3 for SSTF Test SRT 3 Run 26.

In the following sections, first the "parallel channel effects" is discussed for a typical BWR/4 LOCA, then relationship between PCT bifurcation and "parallel channel effects" is investigated.

<u>"Parallel Channel Effects" in a Typical BWR/4 LOCA</u>

The similar "parallel channel effects" was also observed in the BWR/4 LOCA analysis, as described in LTR Section 6.4 in more detail and in Section 8.1.2.2. [[

]]

PCT Bifurcation

The illustrations for PCT bifurcations are shown initially in LTR Figure 6.4-7 and Figure 6.4-8, and recently in the response to RAI-6 in Reference R95-2 (Figure R6-3) for a BWR/4 LOCA. Further investigation of the RAI-6 response calculation in Figure R6-3 was made below, [[

]]

In response to Round 1 RAIs, RAI-6 in particular for BWR/4, GEH has effectively reduced the amount of variability [[

]]

In summary, [[

]]

References

R95-1 NUREG/CR-2566, "BWR Refill-Reflood Program Task 4.4 – CCFL/Refill System Effects Tests (30 Sector) Evaluation of Parallel Channel Phenomena," March 1982.

R95-2 Letter, James F. Harrison (GEH) to NRC Document Control Desk, "Response to Request for Additional Information Re: GE-Hitachi Nuclear Energy Americas Topical Report (TR) NEDE-33005P, Revision 0, "TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant-Accident Analyses for BWR/2-6" (TAC No. ME5405)," MFN 14-064, October 7, 2014.

LTR Impact

The following in LTR Section 8.1.2.2 (the last paragraph) will be modified in response to this RAI.

Original

It was found for small breaks that there was a stronger PCT sensitivity to small perturbations in some of the TRACG model parameters than might be expected a priori. Detailed investigation identified the physical mechanism causing this behavior. When the core begins to drain after the effects of the depressurization have abated and the core pressure drop is low, some of the channels assume a cocurrent upflow mode while others assume a countercurrent flow mode. This behavior is consistent with experimental observations of both these flow modes in the SSTF facility [2]. As was also observed in the SSTF facility, it is possible for bundles to make a transition from one mode to the other during the transient. This behavior may be characterized as bi-stable. Small perturbations to the initial conditions can result in the hot channels changing mode. [[

]]

Revised

It was found for small breaks that there was a stronger PCT sensitivity to small perturbations in some of the TRACG model parameters than might be expected a priori. Detailed investigation identified the physical mechanism causing this behavior. When the core begins to drain after the effects of the depressurization have abated and the core pressure drop is low, some of the channels assume a cocurrent upflow mode while others assume a countercurrent flow mode. This behavior is consistent with experimental observations of both these flow modes in the SSTF facility [2]. ~~As was also observed in the SSTF facility, it is possible for bundles to make a transition from one mode to the other during the transient. This behavior may be characterized as bi-stable. Small perturbations to the initial conditions can result in the hot channels changing mode.~~ ~~[[~~

~~]]~~

[[

]]

Figure R95-1 Liquid and Steam Velocities at Hot Channel SEO (Channel Inlet) for a Case in Figure R6-3 in Reference 2. BWR/4 Suction Line Break of 0.67 ft^2

[[

]]

Figure R95-2 Hot Channel PCT Traces for Two Representative Cases from Reference R95-2 Figure R6-3 Run. BWR/4 Suction Line Break of 0.67 ft^2

[[

]]

Figure R95-3 Liquid and Steam Velocities at Hot Channel SEO (Channel Inlet). BWR/4 Suction Line Break of 0.67 ft^2

[[

]]

Figure R95-4 Void Fraction at Hot Channel Inlet. BWR/4 Suction Line Break of 0.67 ft^2

[[

]]

Figure R95-5 Liquid and Steam Velocities at Hot Channel Node 19. BWR/4 Suction Line Break of 0.67ft^2

[[

]]

Figure R95-6 Void Fraction at Hot Channel Cell 19. BWR/4 Suction Line Break of 0.67 ft^2

SNPB RAI-96

According to Figure 8.3-35 of NEDE-33005P, [[

]] Please provide an evaluation of this result, considering whether the value of this standard deviation is an appropriate indicator of the total uncertainty associated with this result. Consider, in particular, that the standard deviations associated with other break sizes on the same spectrum are much greater.

RAI-96 Response

By reviewing the results of the BWR/2 break spectrum and statistical analyses and similar results for the BWR/4 and the BWR/6 in the LTR, it has been determined that the results for BWR/2 statistical analysis as shown in LTR Figure 8.3-35 for a small discharge break is reasonable.

The detailed discussion of the BWR/2 recirculation line break spectrum is presented in LTR Section 8.3.5.1. The break spectrum response as shown in Figure 8.3-23 for a BWR/2 is the result of [[

]] See LTR Section 8.1.5 for BWR/4 and Section 8.2.4 for BWR/6 for break spectrum discussions.

Because different break sizes affect the relative importance of different phenomena affecting PCT results, statistical results and hence standard deviations obtained from these analyses also vary for different break sizes. A summary of the statistical result for a representative BWR/2 is shown in Figure 8.3-41 of the LTR. As apparent from this plot, the standard deviations vary with different break sizes.

Similar results are also observed for a BWR/4 in Figure 8.1-29 and for a BWR/6 in Figure 8.2-18.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-97

Based on the NRC staff review, NEDE-33005P appears to contain little, if any, discussion on changes in fuel pellet geometry, integrity, and location following fuel cladding ballooning and rupture. Please describe and justify the analytic treatment of these phenomena, with due consideration for available experimental data, and the specific results predicted using the TRACG-LOCA EM, such as fuel and cladding temperatures, fuel rod burnup, extent of cladding deformation, and time of rupture.

RAI-97 Response

The models addressing the effects of fuel pellet and cladding geometry changes are presented in the TRACG Model Description LTR (Reference R97-1) and are considered to be part of the methodology. Particularly, the models describing the effects of geometry changes on fuel pellet gap conductance, including the effects on gap gas conductance heat transfer, fuel-cladding contact heat transfer, gap size calculation, and fuel and cladding thermal expansion, are given in Section 7.5.2 of Reference R97-1.

Section 7.5.3 of Reference R97-1 presents the swelling and the cladding perforation model. The TRACG model is adapted from the previously approved LOCA evaluation model used in SAFER and presented in Reference R97-2. In the model, the fuel rod cladding hoop stress is calculated based on the pressure difference between the fuel rod internal gas pressure and the external coolant pressure during the transient. Initial values for the rod internal pressure and plenum volume, and fission gas parameters are calculated according to the PRIME fuel performance model (Reference R97-3). The effects of thermal conductivity degradation are accounted for in TRACG consistent with the approved PRIME model (Reference R97-3). The change in rod internal pressure during the transient is calculated [[

]]

The transient fuel rod cladding hoop stress is used with the perforation model to determine the onset of cladding plastic yielding and fuel rod perforation. [[

]]

This modeling of plastic strain is also the same as the SAFER model presented in Reference R97-2. In connection with the SAFER code, the NRC reviewed and found the model acceptable as it does not underestimate the incidence of rupture based on applicable data including those data reported in NUREG 0630 (Reference R97-4).

If a fuel rod perforation occurs, the gap conductance is adjusted to reflect the presence of steam and hydrogen rather than fission gases inside the fuel rod. The oxidation and buildup of an oxide layer on the cladding inside surface is allowed after perforation occurs and also contributes to the heat source in the cladding.

[[

]]

Other effects of the cladding deformation include potential relocation of the fragmented fuel into the ballooned region of the rod. When typical operational bundle powers are considered, as the exposure increases, the reduction in LHGR has more influence than the increase in fuel rod pressure and, therefore, fewer fuel rod perforations are expected at very high exposures (Reference R97-9). However, realistic calculations performed at relatively higher LHGR would also indicate that it is possible to swell the rods at all exposures once the cladding has heated up beyond about 800°C. At the low exposures where higher cladding temperatures are expected, the fuel fragments are too coarse to be axially relocated inside the cladding. The concern with fuel relocation within the fuel rod primarily applies for higher exposures where the fuel fragments are smaller.

[[

]] This conclusion is consistent with earlier research results presented in Reference R97-10.

In summary, the calculated clad swelling combined with conservative fragment packing factor is not sufficiently large to pose a concern of added heat source when the fragmented fuel slumps into the ballooned region. This assertion is also valid for high burnup fuel, [[

]]

References

R97-1 GE Hitachi Nuclear Energy, "TRACG Model Description," NEDE-32176P, Revision 4, January 2008.

R97-2 GE Nuclear Energy, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K," NEDO-20566-P-A, September 1986.

R97-3 Global Nuclear Fuel, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance,” Technical Bases – NEDC-33256P-A, Revision 1, Qualification – NEDC- 33257P-A, Revision 1, and Application Methodology – NEDC-33258P-A, Revision 1, September 2010.

R97-4 Powers, D. A. and Meyer, R. O., “Cladding Swelling and Rupture Models for LOCA Analysis,” NUREG-0630, U.S. Nuclear Regulatory Commission, 1980.

R97-5 Ihle, P. and Rust, L. “FEBA – Flooding Experiments with Blocked Arrays, Evaluation Report,” KfK-3657, 1984.

R97-6 Erbacher, F. J., Neitzel, H. J., Wiehr, K., “Cladding Deformation and Emergency Core Cooling of a Pressurized Water Reactor in a LOCA – Summary Description of the REBEKA Program,” KfK-4781, August 1990.

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LTR Impact

No changes to the LTR are proposed as the result of this RAI response.

SNPB RAI-98

Provide additional information to describe the interrelationship among the interfacial shear, entrainment, and wall friction models in TRACG. Particularly, address the qualification of the code to predict two-phase flow and heat transfer behavior in fuel channels at high-void, low pressure conditions that would not otherwise be counter-current flow limited. Such conditions may exist in limiting channels at the point when the code predicts the termination of cladding heatup.

RAI-98 Response

As discussed in Reference R98-1 (Section 3.0), two-phase, two-fluid models are used in the TRACG code. To close the set of basic equations for two fluid models used in TRACG, a set of constitutive correlations describing interfacial shear and heat transfer, wall friction and heat transfer is needed. The response to RAI-69 addresses the application ranges for the models and correlations in each of these four model groups and also provides some details on how they are implemented in the code.

Calculation of interfacial shear and momentum exchange across the interface is a necessary part of the two-fluid equation system solution. In TRACG, the interfacial shear correlations are based on the set of drift flux correlations from void fraction data available in literature, and are provided for each different flow regime (See the discussion in Reference R98-1 Section 6.1.3 for bubbly/church flow, Section 6.1.4 for annular flow, 6.1.5 for droplet flow, and Section 6.1.6 for annular/droplet flow). The modifications to interfacial shear are also considered for subcooled boiling, counter-current flow limitations and virtual mass. For dispersed annular flow interfacial shear, the drift flux parameters are interpolated between the annular and the droplet drift flux parameters based on the entrainment fraction. The model for determining entrainment is discussed in Section 5.1.2.2 of Reference R98-1. How entrainment is used to interpolate the drift flux parameters is defined in Section 6.1.6 of Reference R98-1.

Like interfacial shear, the wall friction or wall shear is another parameter needed to close the set of basic equations for two fluid models in TRACG. The TRACG wall shear model is discussed in Section 6.2 of Reference R98-1. When in annular flow the wall shear is experienced only by the liquid film on the wall. This liquid film also experiences interfacial shear with the vapor phase and any liquid drops entrained within it. Because TRACG calculates only one liquid velocity, the liquid momentum equation contains both wall and interfacial shear force acting on the liquid.

The modified Ishii correlation is used for entrainment prediction in TRACG, as discussed in Reference R98-1 Section 5.1.2.2. The entrainment will affect the interfacial shear calculation in the annular/dispersed flow regime, and also affect the interfacial heat transfer in this regime by affecting the interfacial heat transfer area (See the discussion in Reference R98-1 Section 6.5 for interfacial heat transfer).

For two phase flow in a typical BWR, all those parameters (interfacial shear, wall shear, entrainment and interfacial heat transfer and wall heat transfer) are playing a role together in an entangled way in determining the important parameters for two phase flow, such as pressure drop, void fraction and heat transfer. Qualification of interfacial shear model in TRACG is assessed by examining the capability of TRACG to predict void fraction data, which is presented in Reference R98-2 Section 3.1. TRACG modeling of wall friction is assessed through pressure drop comparisons for the data from the ATLAS test facility in Reference R98-2 Section 3.5. TRACG capability of modeling natural circulation for test data from the FRIGG facility is discussed in Reference R98-2 Section 3.7.

TRACG qualifications are provided in Reference R98-2. Extensive TRACG prediction and test data comparisons have been made. Those comparisons cover the available data from varieties of tests, including separate effects tests, component performance tests, integral system effects tests and BWR plant tests.

The referred scenario in this RAI is observed in the late phase of a typical BWR LOCA, where the RPV has been depressurized, fuel heatup is close to the end and ECCS is cooling the bundle from the top of the bundle for non-jet pump plants (BWR/2s) or from both the top and the bottom of the bundle for jet pump BWRs (BWR/3-6s). Under this condition, [[

]] A summary of TRACG qualification of transition boiling heat transfer was transmitted to the NRC Staff in Reference R98-3 (Starting from Page 177). The most relevant TRACG qualifications to predict two-phase flow and heat transfer behavior in fuel channels at high-void, low pressure conditions [[

]] most closely related to the flow conditions in the RAI are briefly discussed. The detailed discussions of those test facilities and the test events and the results can be found in Reference R98-2. Additional information regarding CSHT and FIST can be found in the LTR and in the response to RAI-92.

- **Core Spray Heat Transfer (CSHT):** These tests were performed at ambient pressure. The bundle is only cooled from the liquid flowing down. This scenario represents the LOCA scenario in the RAI, and occurs in large break LOCA for non-jet pump plants. The detailed description of this test can be found in Reference R98-2 Section 3.2.2. [[
]] Additional

information for CSHT can be found in the response to RAI-92. The comparisons between the test and TRACG predictions showed that TRACG predictions are [[
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- **Full Integral Simulation Test (FIST):** The FIST facility was an integral single-bundle system scaled from a BWR/6-218 Standard Plant. It was capable of simulating full power steady-state operation and real time LOCA and operational transients. The detailed description for FIST can be found in Section 5.2 of Reference R98-2. [[
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- **Two-Loop Test Apparatus (TLTA):** TLTA large break test provided integral system LOCA response data in a scaled BWR facility. The detailed description of this TLTA event can be found in Section 5.1.2 of Reference R98-2, which is similar to the scenario referred in this RAI. The TRACG simulation of TLTA large break test shows [[
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- **ROSA-III Test Facility:** The ROSA-III test facility is a volumetrically scaled (1/424) BWR system with an electrically heated core simulator. It was designed to study the response of the primary system, core and ECCS during a postulated LOCA. The detailed description of the ROSA-III test facility can be found in Section 5.4 of Reference R98-2, which is similar to the scenario referred in this RAI. The comparisons between the ROSA-III test data and the TRACG predictions can be found in Reference R98-2 Section 5.4.

References

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R98-2 GE Hitachi Nuclear Energy, "TRACG Qualification," NEDE-32177P, Revision 3, August 2007.

R98-3 Letter, J. F. Harrison (GEH) to Document Control Desk (NRC), "Final Presentation for ACRS Thermal-Hydraulic Phenomena Subcommittee Meeting on September 21, 2015 and Final Presentation and Requested Information for NRC Audit Related to the Peach Bottom Units 2 and 3 License Amendment Request for MELLLA+, August 31, 2015 to September 2, 2015," MFN 15-078, September 25, 2015.

LTR Impact

No changes to the LTR are proposed as the result of this RAI response.