September 6, 2007

Mr. Robert E. Brown
Senior Vice President, Regulatory Affairs
GE-Hitachi Nuclear Energy Americas, LLC
3901 Castle Hayne Rd MC A-45
Wilmington NC 28401

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 106 RELATED TO ESBWR DESIGN CERTIFICATION APPLICATION

Dear Mr. Brown:

By letter dated August 24, 2005, GE-Hitachi Nuclear Energy Americas, LLC (GEH) submitted an application for final design approval and standard design certification of the economic simplified boiling water reactor (ESBWR) standard plant design pursuant to 10 CFR Part 52. The Nuclear Regulatory Commission (NRC) staff is performing a detailed review of this application to enable the staff to reach a conclusion on the safety of the proposed design.

The NRC staff has identified that additional information is needed to continue portions of the review. The staff's request for additional information (RAI) is contained in the enclosure to this letter.

Pursuant to 10 CFR 2.390, we have determined that the enclosed RAI contains proprietary information. We have prepared a nonproprietary version of the RAI (Enclosure 1) that does not contain proprietary information. The proprietary information is indicated in brackets and underlined in Enclosure 2. We will delay placing this document in the public document room for a period of ten (10) working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any additional information in the enclosure is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.390 before the public release date.

To support the review schedule, you are requested to provide the requested additional information within 45 days of the date of this letter.

If you have any questions or comments concerning this matter, you may contact me at 301-415-6715 or bmb2@nrc.gov or you may contact Amy Cubbage at (301) 415-2875 or aec@nrc.gov.

Sincerely,

/RA/

Bruce Bavol, Project Manager
ESBWR/ABWR Projects Branch 1
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-010

Enclosure: 1. Request for Additional Information (Non-Proprietary)
2. Request for Additional Information (Proprietary)

cc: See next page (w/o enclosure 2)

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RAI Number	Reviewer	Question Summary	Full Text
4.2-12 Supplemental Request 2 (MFN-06-492 Supplement 1 dated June 20, 2007)	Yarsky, P	Accounting for uncertainties in the LHGR limit	Parts 1 and 2 will remain open items until these issues are acceptably resolved by RAI 4.3-2. Part 6: Provide any relevant data that would be indicative of discharge exposure. Namely, provide the core thermal power level, core size, and cycle duration. Using any additional relevant information, provide an estimate of the average cycle exposure. Alternatively qualitatively assess any design features of K5 relative to the ESBWR to determine if the discharge exposures are expected to be significantly different. Part 8: Please provide greater clarification of what is meant by the "interim methodology." Does this interim methodology correspond to the interim methodology for expanded operating domain BWRs? Part 10: The insight that the staff needs is to understand the impact on predicted power distributions for each adaption technique. Additionally, the staff was not aware that the uncertainty analysis for GT instrumentation is predicated on the [[ ]] methodology as opposed to the proposed methodology for the ESBWR (PANAC11). The ESBWR uncertainty analysis, it appears to the staff, may depend on the core simulator and the adaption technique employed. This adaption technique will also depend on the number of AFIPs or other [[ ]] methods. Since the information regarding the K5 reactor is sparse, the core monitoring software was different, the number of AFIPs proposed for ESBWR and those employed at K5 are different, and no final adaption technique has been proposed, the staff does not have sufficient information regarding the numerical

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			uncertainty analysis to make a determination regarding the applicability of the K5 data to the proposed ESBWR application. To provide insight into the effects of adaption on power distribution uncertainty, please provide an analysis using a relevant reactor plant from the experience database. Using purely predictive methods (no adaption) perform a core follow analysis for a relevant (high power density, large core) reactor plant. The plant and cycle selected for reanalysis should be challenging from a reactor power distribution standpoint. [[ ]] Produce a MOC and an EOC radial power map (axially integrated four bundle power) and axial power shape curve. Please provide these curves in figures that are substantially similar in format to Figures 27-1 through 27-68 of MFN-05-029. Using LPRM adaption, perform the same core follow analysis and produce a MOC and an EOC radial power map and axial power shape curve. Provide additional figures using TIP adaption. Specify whether absolute or shape adaption is used. When an adaption technique is finalized for the ESBWR, [[ ]] readings based on local TIP readings, perform a cycle follow analysis and associated radial and axial power distributions for the same plant using [[ ]] adaption with an arrangement that is similar to the ESBWR (i.e. [[ ]] instruments per string with similar spatial arrangement). The staff understands that this will not help assess the [[ ]] uncertainty, but it will provide a quantitative comparison of core monitoring performance using discrete vs. continuous adaption. Comment on the differences in the radial and axial power distributions based on each adaption technique. Please also provide quantitative comments in regards to the expected uncertainty when using PANAC11 methods (including updates to TGBLA06) relative to the uncertainty analysis that is based on [[ ]] methods.

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			Part 11: The staff requires additional information in regards to the uncertainty analysis in order to determine the acceptability of the design to ensure SAFDLs are not exceeded. The OLMCPR and the MLHGR limits are predicated on uncertainty assessments (a demonstration that the pin power uncertainty is less than [[ ]] for the latter). Part 12: Provide the core thermal power and core flows for the other reactors described in NEDC-33197P, namely [[ ]] for the times of the respective tests. Compare the power to flow ratios for these plants during the tests to that for the ESBWR. Part 16: The ESBWR uncertainty analysis, it appears to the staff, may depend on the adaption technique employed. This adaption technique will also depend on the number of AFIPs or other [[ ]] methods. If the adaption technique is not finalized, provide separate uncertainty analyses for each available technique, or each unique available combination of measurements, calibrations, [[ ]], intervals, and adaption techniques. For example using different adaption techniques, or [[ ]] for the [[ ]] cycle follow would generate different values for the [[ ]]. Part 17: Update the NEDC-33197P topical report to include an appendix that summarizes the available techniques described in the supplemental information request Part 16. In the appendix describe the uncertainty assessment methods that are used to obtain uncertainties which are used in downstream safety and operating limit determinations based on each available technique.

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			Part 19: The response states that the adaption technique is still under development. If a single adaption technique (as opposed to many alternatives) is developed, provide the information requested in Parts 16 and 17 for only that one technique. Part 20: The staff does not find the response acceptable. If the adaption technique is based on discrete axial signals, perhaps 4 LPRM signals or [[__________]] GT signals, the axial power shape uncertainty would likely be a function of the resolution provided by those signals. [[ ]]. Once a single, or perhaps several alternative adaption techniques, are selected, provide a basis for each technique that the number of GTs is sufficient such that the uncertainty analysis results are applicable even if there are power shapes other than cosine, bottom-, or top- peaked. Part 22: Provide the results of GE14 corroborative MCNP/[[ ]] analyses that were performed for a representative [[ ]] lattice. Include at least one case that considered a spacer. Part 25: See the supplemental request in Part 17.
4.3-2 Supplemental Request 02 (MFN-7-350 Supplement 3 dated July 15, 2007)	Yarsky, P	Provide the experience database in tabulated form, pertinent to expected operation of the ESBWR, and operation with high exit void	A. Confirm that the [[ ]] peak rod power uncertainty bounds not only those lattices in the equilibrium ESBWR core, but also those in the initial core. B. The response indicates that a SLMCPR analysis was performed for the ESBWR. Was this SLMCPR analysis performed according to the approved SLMCPR methodology for operating reactors? If so, please provide this analysis. C. As discussed in the staff's RAI 4.2-12 and MFN-05-029, the uncertainty in gamma instrument measurement increases with increasing power to flow ratios.

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		fractions	1. The ESBWR power to flow ratio is substantially higher than that for [[ ]]. Describe what approach is being taken to account for this phenomenon in the overall assessment of power distribution uncertainties. In other words the determination of the [[ ]] and may not be representative of a similar quantity determined for conditions of operation similar to the ESBWR. 2. The response to RAI 4.4-39 S01 [[ ]], comment on the effect of bypass voiding due to high power to flow ratios on the sensitivity of the GT and the ability of the methodology as proposed to account for changes in sensitivity arising from bypass voiding. Please consider effects such as heat transfer from the jacket tube to the two-phase mixture (given the predicted bypass flow patterns) as well as gamma attenuation and streaming. D. The foot note in Table 9-2 states that more data is required for application. Explain why [[ ]] results in Table 7-2 were not combined with the [[ ]] data in Tables 7-3 and 7-4 to assess this uncertainty. The information in Table 9-8 seems to indicate that the [[ ]] data would be applicable. E. How are the [[ ]] uncertainties in Table 4.3-2S01-2 weighted to determine the total estimated uncertainty per GT string? F. The GT strings used to assess the bundle power uncertainties each include [[ ]] instruments per string. The ESBWR design includes [[ ]] instruments per string. The staff does not understand how the same uncertainties will apply if there are [[ ]] instruments. In response to RAI 7.5-58 (MFN-07-162) the response states that it is “not realistic to conclude that the uncertainty is not dependent on the number of GT sensors per string… Table 9-8 indicates that having fewer GT sensors per string results in smaller uncertainties, this result arose only because the study was not realistic and based only on simulated GT readings. In practice, the

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			uncertainty will be larger with fewer GT sensors per string.” This statement does not appear to be consistent with the numerical values provided in the uncertainty analysis in the response to RAI 4.3-2. Please update the uncertainty analysis to include a term that addresses the [[ ]] sensors. If the basis for determining this uncertainty is provided in a separate RAI response, please provide a specific reference. G. The response indicates that the ESBWR generic R-factor uncertainty was determined in a manner that is conservative relative to the prescription in the interim methods. Please provide an update to NEDC-33239P that confirms that the R-factor uncertainty is consistent with ESBWR pin power peaking and power allocation uncertainties as determined in a manner consistent with the prescription in the approved interim methods (NEDC-33173P-A). The staff understands that GE will supplement this topical report with additional data for review to support the historical R-uncertainty analysis inputs. The update may make reference the most recently approved version of NEDC-33173P-A.

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4.4-39 Supplemental Request 02 (MFN-7-350 Supplement 3 dated July 15, 2007)	Yarsky, P	Characterize core outlet pressure distribution	The staff disagrees with the applicant’s assertion that the TRACG and PANACEA calculations are independent based on information provided in the response to RAI 21.6-85. It is the staff understanding that [[ ]]. Since the thermal hydraulic and neutronic solutions are tightly coupled by the void reactivity coefficient it is not unexpected that TRACG would converge on a thermal hydraulic solution that matches the axial void profile predicted by PANACEA. However, as an inherent artifact of the methodology the power distribution will always be calculated predicated on a [[ ]] the code system proposed in the original RAI response as an “independent” verification approach will artificially compensate through the nuclear feedback to the thermal hydraulic solution and therefore is not considered independent by the staff. The staff questions the validity of the assumption for the following reasons: 1. The high power density of the ESBWR core will result in bypass voiding due to gamma heating below the TAF that is not insignificant. 2. The chimney partitions block thermal hydraulic communication above the top guide between super bundles. Therefore, the staff requests that the applicant perform an analysis to determine the core outlet pressure distribution using an independent verification approach.

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6.2-154, Supplement No. 1, MFN 07-313, 6/22/07	Haider S	Include the RAI response in the DCD	The information provided in this response is adequate and necessary to support the basis for a reasonable assurance finding. Thus, please update DCD Tier 2 to include information provided in response to RAI 6.2-154.
6.2-179	Goel, R	Discuss compliance with TMI Action Plan Item II.E.4	The governing regulation for TMI Action Plan Item II.E.4.4, Containment Purging During Reactor Operation, is 10 CFR 50.34(f)(2)(xv), which states: Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4) The DCD entry on this generic issue, in Table 1A-1, “TMI Action Plan Items,” simply asserts that the ESBWR design complies with these requirements, without explanation or justification. What follows is a discussion of the bases for the generic issue. The first requirement of the regulation refers to a situation that generally does not occur in a plant with an inerted containment atmosphere, which is unwarranted or excessive containment purging. The NRC established this generic issue because it had found that some (non-inerted) plants were purging/venting their containments for sizable fractions of the plant’s operating time, or even continuously. The NRC recognized that an open purge/vent line constitutes a sizable hole in the containment boundary, which is intrinsically a less safe condition than having all purge/vent valves closed, in case an accident occurred. One legitimate reason for purging while the reactor is operating is to reduce the concentration of airborne radioactive material in the containment atmosphere, which would reduce personnel occupational exposure for personnel

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			who enter containment. The regulation, then, calls for minimizing purging time, consistent with ALARA principles for occupational exposure. However, personnel do not enter containments while they are inerted, so there is no need to purge for this reason. In general, plants with inerted containment will naturally minimize purge/vent time (except when inerting or de-inerting) because of the cost of the nitrogen gas needed to replace that which is expelled from containment. Also, as mentioned before, personnel exposure during containment entries is not a factor. The second requirement of the regulation, to provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions, is explained in more detail in NUREG-0737, item II.E.4.2, subpart (6) and Attachment 1. The staff had found that some purge/vent valves in operating plants, typically butterfly valves, were not capable of closing if a design bases (DB) loss of coolant accident (LOCA) occurred while the valves were open. In a DB LOCA, containment pressure increases so rapidly that the containment atmosphere rushes out through open purge/vent valves before they can begin to close. Some valves were found to be incapable of closing against the aerodynamic forces induced by the rapidly moving gas; in fact, some valves would even be damaged by the transient so that they would be stuck open and incapable of closing again until repaired. The regulation, therefore, requires the applicant to demonstrate, by analysis and/or testing, that the purge/vent valves would be capable of closing under these conditions. An alternative to such demonstration is to assure that purge/vent valves will never be open while the plant is operating, by including a requirement in the Technical Specifications (TS) that they must be locked or sealed closed in Modes 1 through 4, with no exception for even momentary opening of a purge/vent line while in Modes 1 through 4. The ESBWR TS SR 3.6.1.3.1 indicates that the ESBWR purge/vent valves will not be sealed closed.

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			Note that this issue extends beyond the 500 mm (20 in) purge valves covered by TS SR 3.6.1.3.1. Other systems which may purge/vent the containment, regardless of what they are called, must be included. Some or all of the valves in the containment inerting system, for example, will be opened to purge/vent the containment in Modes 1 through 4 and must also be demonstrated to reliably isolate under accident conditions. Provide the following information: A. Provide a discussion in the DCD which presents arguments or justifications to demonstrate compliance with the requirement of 10 CFR 50.34(f)(2)(xv) to provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. B. Provide and demonstrate in the DCD high assurance that the purge system will reliably isolate under accident conditions, or provide TS which require purge/vent valves to be sealed closed in Modes 1 through 4. C. Identify in the DCD all purge/vent valves. This includes all containment isolation valves (CIVs) in lines that perform a purging or venting function - meaning transferring gas between the containment atmosphere and the outside atmosphere. This may include some or all of the CIVs in the containment inerting system, and perhaps others. All purge/vent valves are subject to the requirements of 10 CFR 50.34(f)(2)(xv).
RAI 6.3-18, Supplement No. 2, (MFN 06-241, Supplemental No. 2, dated April 12, 2007)	Thomas G Klein V	GDCS Flow Test	GEHs response did include the proposed ITAAC (items 2 a, b) the as-built flow loss coefficient (K/A^2) for the GDCS injection and the equalizing line. However, the proposed ITAAC does not include the flow loss coefficient values assumed in the TRACG analyses. Include the flow loss coefficient values assumed in the TRACG analyses in the Acceptance Criteria.

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RAI 6.3-24, Supplemental No. 2, (MFN 06-241, Supplemental No. 2, dated April 12, 2007)	Thomas G	GDCS Equipment Qualification	GEHs response stated that the DCD, Tier 1, Subsection 2.4.2, Design Description, includes a statement that “All GDCS safety-related components are qualified to withstand the harsh environments postulated for design basis accidents.” This requirement is not included in the ITAAC Table 2.4.2-1. Include the requirement in the ITAAC Table 2.4.2-1 for verification.
RAI 6.4-18	Forrest E	Correct the DCD	A. DCD, Tier 2, Revision 3, Table 1.9-9 states for SRP Section 9.4.5 “The engineered safety features described in Chapter 6 do not require a separate ventilation system. This section is not applicable to ESBWR.” Please make the appropriate correction in the DCD to account for the addition of the EFU system in DCD revision 3. B. DCD, Tier 2, Revision 3, Table 1.9-6 states that SRP Section 6.5.1 is not applicable to the ESBWR. Please make the appropriate correction in the DCD to account for the addition of the EFU system in DCD revision 3. C. DCD, Tier 2, Revision 3, Table 1.9-20 states that SRP section 6.5.1 is not applicable to the EWBWR and comments that there is no standby gas treatment. Please make the appropriate correction in the DCD to account for the addition of the EFU system in DCD revision 3.
9.5-44 Supplement No. 1 (MFN 07-319, June 18, 2007)	Radlinski, R	Clarify in the DCD that the post-fire, safe shutdown circuit analyses will be developed by the licensee	The GE response to RAI 15.5-3 states that "The ESBWR post-fire, safe shutdown circuit analyses have not been developed at this time. These analyses will be developed later in the project life cycle as part of the plant specific fire protection program." Consequently, the DCD should be revised to state that the post-fire, safe shutdown circuit analyses will be developed by the licensee as part of the plant specific fire protection program.

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9.5-45 Supplement No. 2 & 9.5-46 Supplement No. 2 (MFN 07-260, Supplement No. 2, August 17, 2007)	Radlinski, R	Provide information in the fire hazard analysis consistent with RG 1.189 and NFPA 804.	Contrary to GE's response that the fire hazard analysis provided in the ESBWR design certification document provides all of the information needed for a fire hazards analysis in accordance with NFPA 804 and Regulatory Guide 1.189, the following information listed in these two guides has not been provided to the extent described: RG 1.189 1. Amounts, types, configurations and locations of flammable and combustible materials – in situ and transient. 2. Layout and configuration of structures, systems and components important to safety. 3. Accessibility of plant areas for manual fire fighting, location and type of manual fire fighting equipment 4. Lack of adequate access or smoke removal facilities that impede plant operations or fire extinguishment in plant areas important to safety. NFPA 804 1. All in situ combustibles and flammable materials and their configurations should be identified. Where in situ combustibles present an exposure to nuclear safety-related systems and components, they should be uniquely identified. 2. Physical construction and layout of the buildings and equipment. 3. Description and location of any equipment necessary to ensure a safe shutdown, including cabling. 4. Analysis of smoke control system and the impact smoke can have on nuclear safety and operation for each area. 5. Analysis of the emergency planning and coordination requirements

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			necessary for effective loss control, including any necessary compensatory measures to compensate for the failure or inoperability of any active or passive fire protection system or feature. As noted in the GE response to RAI 9.5-46, RG 1.206 does acknowledge that this information may not be available even at the COL stage, but the RG also stipulates that if the information is not available, the applicant should justify the inability to provide the unavailable information in the COL application, and furnish details describing implementation plans, milestones and sequences and/or ITAAC or commitments for developing, completing, and submitting this information during the construction period, prior to fuel receipt at site. This should be identified in the DCD as a COL action item. While this level of information may not be important in areas that provide complete 3-hour fire barrier separation between redundant post-fire safe-shutdown trains, as a minimum, the information should be developed and documented by the applicant for areas where full compliance with the criteria for enhanced fire protection is not feasible. In particular, the level of information described above should be provided for the main control room and adjacent rooms where exceptions have been taken to the regulatory guidance for fire protection, as well as for the other "Special Cases" described in DCD Section 9A.6.
21.6-65, Supplemental No. 1, (MFN-07-347, dated June 21, 2007)	Klein, V	Selection of channels to evaluate ΔCPR/ICPR	The original RAI requested GEH to provide additional information about the TRACG nodalization used to model ESBWR anticipated operational occurrences (AOO) and infrequent events (IEs). The staff requests the following additional information to complete its review of this portion of the ESBWR design certification: How are the channels selected for evaluating the maximum ΔCPR/ICPR that is used to determine the OLMCPR? Do you use the hot channel every time? Or do you take the maximum of all the channel groups? The staff is concerned for cold water injection events where although the Ring 3 channels (peripheral channels) do not have a hot channel, it is possible that these channel groups may experience the highest ΔCPR/ICPR.

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21.6-79, Supplemental No. 1, (MFN 07-256, dated May 17, 2007)	Klein, V	Ranking of minimum stable film boiling temperature	The original RAI requested GEH to justify the choice of the minimum stable film boiling temperature model for ESBWR events. In GEH's response they state that this model is not important for ESBWR events because, with the exception of anticipated transients without a scram (ATWS), they do not enter film boiling. And even for ATWS, GEH states that the minimum stable film boiling temperature is only used to determine when the core will quench and does not have any effect on the value of the maximum peak cladding temperature. Please explain why this parameter was ranked high in the TRACG application for BWR/2-6 AOOs, PIRT C13 (see Reference), and why this explanation is not applicable for ESBWR. Reference: NEDE-32906P-A, Rev. 2, MFN 06-046, *TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis,* February 28, 2006 (ADAMS Accession No. ML060530571 and ML060530575)
21.6-82, Supplemental No. 1, (MFN-07-309, dated June 8, 2007)	Yarsky, P Klein, V	Effects of transient Xenon on ESBWR start-up	The original RAI requested details of the transient xenon capability in TRACG. GEH responded that xenon concentration remains constant and is not updated during TRACG transients. The staff agrees with GEH that actual xenon concentrations would have insignificant impact on anticipated operational occurrences (AOO) and anticipated transients without a scram (ATWS) calculations. However during start-up, which takes place over the course of hours, the staff believes that xenon will have an impact on calculations. The staff is concerned about stability during start-up. A condition put upon the approval of TRACG for predicting stability margins for ESBWR (see Reference) was: "The ascension to full power, as with all other transient xenon conditions, will be analyzed using the PANACEA transient xenon option and reviewed at the design certification stage." Since TRACG does not have transient xenon capability, please provide additional information to address the effects of xenon on ESBWR start-up. Reference: Safety Evaluation Report Regarding the Application of General Electric's Topical Report, *TRACG Application for ESBWR Stability Analysis*, NEDE-33083P, Supplement 1 (TAC MC3288) (ADAMS Accession Nos. ML0608900750 and ML061000463)

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21.6-84, Supplemental No. 1, (MFN-07-309, dated June 8, 2007)	Klein, V	Update to void coefficient bias and uncertainty	The original RAI requested GEH to provide the details about the void coefficient bias and uncertainty used in TRACG. GE stated that the comparison had been updated from PANAC10/TGBLA04 to PANAC11/TGBLA06. Please provide a description of the lattices used in determining the void coefficient bias and uncertainty.
21.6-96, Supplemental No. 1, (MFN-07-348, dated June 21, 2007)	Klein, V	Use of PC versus ALPHA VMS versions of TRACG	The original RAI requested GEH about differences found in the calculated results seen in the ALPHA VMS versus PC versions of TRACG on containment peak pressure. A. The RAI response states that TRACG cannot accurately predict noncondensible gas distributions in general and that a conservative approach was to minimize the long-term pressure response sensitivity to noncondensible concentrations by modifying the input model nodalization to force all the air out of the drywell. This approach may not necessarily be conservative for long-term core cooling calculations where the presence of non-condensibles in the PCCS would degrade the capability of the PCCS to condense steam and return inventory back to the vessel. Provide justification that the treatment of non-condensible gases is conservative with respect to long-term core cooling analyses. B. During a phone call with NRC staff on this RAI, the NRC staff expressed concern that GEH was using an unqualified code version to perform design calculations. GEH staff stated that to address the concern they would provide an appropriate sub-set of the TRACG qualification, as determined by GEH, using the PC version of the code to demonstrate that this version of the code produces reasonably accurate or conservative results. Please provide the qualification.

RAI Number	Reviewer	Question Summary	Full Text
21.6-100, Supplemental No. 1, (MFN-07-348, dated June 21, 2007)	Klein, V	CHAN leakage flows	The original RAI requested additional information about the CHAN leakage model. In response GEH stated a reference. The staff does not have any information on the "GE Design Leakage Flow correlations" except for the name of the reference. A. Please provide the following reference so that the staff can verify the information provided in the RAI response: B.S. Shiralkar and J. R. Ireland, "Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR Appendix K, Amendment No. 5, Backflow Leakage from the Bypass Region for ECCS Calculations," NEDE-20566-5P, GE Proprietary Report, June 1978. B. In addition, please explain how the CWF and CWB are determined to be appropriate for ESBWR.