January 19, 2007

Mr. David Hinds, Manager, ESBWR
General Electric Company
P.O. Box 780, M/C J70
Wilmington, NC 28402-0780

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 85 RELATED TO ESBWR DESIGN CERTIFICATION APPLICATION

Dear Mr. Hinds:

By letter dated August 24, 2005, General Electric Company (GE) 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. This RAI concerns Chapters 5, 6, 7, and 16 of the ESBWR Design Control Document and Chapters 20 and 21 of the staff's development of the ESBWR safety evaluation report.

Chapter 5:	5.3-14
Chapter 6:	6.2-144 through 6.2-153 and 6.3-62 through 6.3-75
Chapter 7:	7.7-6
Chapter 16:	16.2-111
Chapter 20:	20.0-7
Chapter 21:	21.6-95 through 21.6-100

To support the review schedule, you are requested to respond to this RAI by March 2, 2007.

If you have any questions or comments concerning this matter, you may contact me at (301) 415-3207 or saw8@nrc.gov, or Amy Cubbage at (301) 415-2875 or aec@nrc.gov.

Sincerely,

/RA/

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

Docket No. 52-010

Enclosure: As stated

cc: See next page

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ACCESSION NO. ML070080448

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DATE	01/18/2007	01/19/2007

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RAI Number	Reviewer	Question Summary	Full Text
5.3-14	Lois L	Discrepancy of applied and approved methods	DCD Tier 2, Revision 2, Section 4.1.4.5 states that the cross sections are prepared with I/E spectrum but do not include self-shielding factors. The methodology referenced in NEDC-32983P-A states in Section 2.1.6 that the cross section library is used in performing the resonance self shielding. Is the methodology not applied as described in the approved document and if so what is the basis?
6.2-144	Wagage H	Justification for the containment back pressure used for determining the minimum RPV water level	DCD, Tier 2, Revision 2, Section 6.3 assumed the availability of the containment back pressure in determining the minimum water level in the reactor pressure vessel (RPV) following a LOCA. The depressurization of the RPV and thus the initiation of the Gravity Driven Cooling System (GDCS) is dependent on the assumptions used for determining the containment back pressure. However, the analyses are inconsistent with Standard Review Plan (SRP) Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," Revision 2, July 1981 and Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." Although, CSB 6-1 was developed to evaluate the performance of emergency core cooling system of a pressurized water reactor, most of its guidance is also applicable to ESBWR for determining the performance of the GDCS. Specifically, the input information for the model, active heat sinks (e.g., Fuel and Auxiliary Pools Cooling System operating on drywell spray mode), and passive heat sinks affect the containment back pressure. Please justify the containment back pressure used for determining the minimum RPV water level considering Branch Technical Position CSB 6-1.
6.2-145	Wagage H	Acceptance criterion for the drywell to wetwell bypass leakage tests of 0.1 cm^2 ($A/\sqrt{K}$)	DCD, Tier 2, Revision 2, Section 6.2.1.1.5.1 states that the bounding design basis calculation assumed a bypass leakage of 1 cm^2 ($A/\sqrt{K}$). This value is significantly lower than the design capacities of Mark I, II, and III containments: 18.6 , 46.5, 929 cm^2 ($A/\sqrt{K}$), respectively (SRP Section 6.2.1.1.C, Revision 6, August 1984). DCD, Tier 2, Revision 2, Section 6.2.1.1.5.4.3 states that the acceptance criterion for the bypass leakage area for the leakage tests will be 10% of 1 cm^2 ($A/\sqrt{K}$) (i.e. 0.1 cm^2 ($A/\sqrt{K}$)). Please explain why you believe the plants will be able to meet and maintain the bypass leakage at such a low value.

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6.2-146	Wagage H	Discrepancy between Section 6.2.1.1.5.4.3 and SR 3.6.1.1.2	DCD, Tier 2, Revision 2, Section 6.2.1.1.5.4.3 states that the acceptance criterion for the bypass leakage area for the leakage tests will be 10% of 1 cm^2 (A/√K) (i.e. 0.1 cm^2 (A/√K)). Surveillance Requirement 3.6.1.1.2 given in DCD, Tier 2, Revision 2, Chapter 16 is to "[v]erify drywell to wetwell bypass leakage is less than 1 cm^2 (A/√K)." Please correct this discrepancy.
6.2-147	Wagage H	Adding the limiting DW to WW leakage size to DCD	In response to NRC RAI 6.2-12, in letter MFN 06-159, dated June 5, 2006, GE stated that a sensitivity analysis showed that the peak drywell pressure of a feedwater line break accident would approach the design pressure of 45 psig at 72 hours after the pipe break if the leakage size were increased to A/√K = 100 cm^2. Please add this information to the DCD.
6.2-148	Wagage H	Information of the isolation valve for vacuum breaker	DCD, Tier 2, Revision 2, Section 6.2.1.1.2 states that "On the upstream side of the vacuum breaker, a DC solenoid operated isolation valve designed to fail-close is provided." Please state what type of a valve it is and how the fail-close function is provided.
6.2-149	Wagage H	Loss coefficients in Table 6.2-12 not clear	Please provide the following for the staff to understand the information given in DCD, Tier 2, Revision 2, Table 6.2-12: • 8^{th} column: define "t" given for the column heading and explain how you determined it • 11^{th} column: explain how you determined the analysis loss coefficient.
6.2-150	Wagage H	Assumptions used for calculating the limiting drywell spray flowrate	DCD, Tier 2, Revision 2, Section 6.2.1.1.4 states that "In order to prevent excessive negative pressure the drywell spray flow rate must be less than 227 m^3/hr (1000 gpm)." Please state the assumptions used to calculate this spray flow rate, including the timing of spray initiation.
6.2-151	Wagage H	Additional conservatism not stated	DCD, Tier 2, Revision 2, Section 6.2.1.3 states that "additional conservatism is included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA." Please state the additional conservatism included in the DCD.

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6.2-152	Wagage H	The volume of the GDCS pools are not added to the volume of the drywell	DCD, Tier 2, Revision 2, Section 6.2.1.1.10.2 state that "Once the [gravity driven cooling system] GDCS pools are drained, the total volume of the GDCS pools are added to the volume of the drywell airspace." The Staff believes that adding volume to the drywell airspace is not possible because the water removed from the GDCS pools would occupy the drywell volume. Please explain this statement or correct the DCD.
6.2-153	Wagage H	Typographical errors	Please correct the following typographical errors in DCD, Tier 2, Revision 2: • In Section 6.2.3.3: "Calculated pressure responses have been considered in order to define the peak pressure, of the RB compartments, for structural design purposes" to "Calculated pressure responses have been considered in order to define the peak pressure of the RB compartments for structural design purposes" • On Table 6.2-3: "Suppression Pool Depth at Low Water Level 5.4 m (17.7)" to "Suppression Pool Depth at Low Water Level 5.4 m (17.7 ft)" • On Table 6.2-9: "Automatic Depression System (ADS)" to "Automatic Depressurization System (ADS)"
6.3-62	Klein V Yarsky P	Justify the ESBWR Shutdown Power	DCD Tier 2, Revision 2, Section 6.2.1.3 states that "Calculations of the energy available for release from the above sources are done in general accordance with the requirements of 10 CFR 50, Appendix K, paragraph I.A. However, additional conservatism is included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA." Please provide: (A) A list of the conservative assumptions applied to the calculation of the decay heat for the ESBWR. Where applicable, quantify these conservative assumptions (e.g. if a reactor scram delay of 1 second is conservatively assumed, the additional heat load on the containment is given by the integrated reactor power over this interval). If an increase in core average exposure is applied to ensure conservatism in the decay heat, verify that the plutonium fission, and hence fractional decay heat contribution, was not artificially increased by providing the irradiation time and exposure assumptions for the analysis. (B) A justification of the application of the same decay heat curve to both small and large break LOCAs. Particularly, provide an analysis demonstrating that the assumptions used

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			to calculate the fission power during rod insertion and fission power from delayed neutrons are conservative for a full spectrum of LOCAs. If a TRACG analysis is used to demonstrate the conservatism, fully describe the calculation; including the specific decay heat option used, the initiator for the SCRAM signal, the evolution of the core average void fraction, and any assumptions regarding blade insertion time or axial load. (C) A description of the heat sources considered in the decay heat analysis. If an evaluation of the decay heat model after ~100 seconds was performed to ensure that the ANS 1994 standard included sufficient conservatism to neglect the integrated effects of heat from other sources, describe the evaluation process and results. Otherwise, justify not including those sources. (D) A description the distribution of the decay heat in the TRACG core model. Is the decay heat distribution based on the initial flux shape or the initial power shape? (E) A comparison of the ESBWR specific design values for the hydraulic control unit valve deenergization and stroke time, instrument and logic delay time, and control blade insertion time to those values assumed in the calculation of the decay heat curve. (F) An explanation of the reference to the 1979 ANS standard in DCD Tier 2, Revision 2, Table 6.3-11. (G) A section in the next revision to the DCD that describes the assumptions in the shutdown power calculation that ensure that the power is conservative for a full spectrum of LOCAs for the ESBWR.

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6.3-63	Klein V	Vacuum Breaker Isolation valve	DCD, Tier 2, Revision 2, Page 6.2-3 states that: "During a [loss of coolant accident] LOCA, when the vacuum breaker opens and allows the flow of gas from the [wetwell] WW to the [drywell] DW to equalize the DW and WW pressure and subsequently does not completely close as detected by the proximity sensors, a control signal will close the upstream isolation valve to prevent extra bypass leakage due to the opening created by the vacuum breaker and therefore maintain the pressure suppression capability of the containment." How does the control system determine that there is extra leakage due to the vacuum breaker not closing? What is the leakage threshold? What is the time delay signal for the closing of the isolation valve? Given the allowed leakage and time delay of the isolation valve, what is the impact on the passive containment cooling system performance? How does this effect containment pressure and reactor pressure vessel level calculations?
6.3-64	Klein V	Long term core cooling	Show plots of the core level demonstrating that the core remains covered for 72 hours for the limiting break. Justify that the input deck assumptions used for calculating long-term core level are conservative.
6.3-65	Klein V	SLCS Line Break	Show the elevation, diameter and maximum break area of the standby liquid control system (SLCS) injection line (which may be included in response to RAI 6.3-47). Evaluate the consequences of a break in the SLCS line with the worst single failure.
6.3-66	Klein V	Include IC Heat Removal and HCU as ECCS Systems in DCD	Revise DCD, Tier 2, Chapter 6.3 to include a statements that the loss of coolant accident reactor pressure vessel level analyses takes credit for isolation condenser heat removal capacity and hydraulic control unit injection.
6.3-67	Klein V	Changes to GDCS Design Basis Parameters	Explain the changes in DCD, Table 6.3-2 "[gravity driven cooling system] GDCS Design Basis Parameters," from DCD Rev. 1 to DCD Rev. 2. These were not described in the Chapter 6 "Change List" document.
6.3-68	Klein V	Include figures of Reactor Power	Include figures of reactor power vs time for the breaks in Chapter 6.3 of the DCD.

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6.3-69	Klein V	Include figures of Break void fraction	Include figures of void fraction vs time for the break flow for the breaks presented in Chapter 6.3 of the DCD.
6.3-70	Klein V	Update the DCD to state that the LOCA are nominal or bounding cases	Update the DCD to state that the figures represented on pages 6.3-48 through 6.3-79 are nominal or bounding cases.
6.3-71	Klein V	Explain the phenomena occurring around 150-175 seconds into the Feedwater line break	For the feedwater line break with 1 gravity driven cooling system valve failure displayed in Figures 6.3-7a through 6.3-14b of the DCD Rev. 2, please explain what occurs at around 150-175 seconds that causes the change in minium critical power ratio, chimney level, downcomer level, break flow, and isolation condenser steam flow.
6.3-72	Klein V	Correct typos related to time axis of Figures in 6.3	In DCD, Revision 2, the x-axis on Figures 6.3-8a, 6.3-9a, 6.3-16a, 6.3-17a, 6.3-25a is labeled as time in hours. The staff believes this to be a typographical error. If so, please correct these typographical errors in the next revision of the DCD.
6.3-73	Klein V	Explain the phenomena occurring around 400 seconds into the Inside Steam line break event	For the inside steam line break with 1 gravity driven cooling system valve failure displayed in the DCD, Revision 2, Figures 6.3-15a through 6.3-22b, please explain what occurs on or before 400 seconds that causes the change in minium critical power ratio and chimney level. Also please explain the small oscillations in drywell pressure seen shortly before this time.

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6.3-74	Klein V	Explain the increase in IC steam flow around 200 seconds for all breaks	All of the breaks displayed in DCD, Revision 2, Section 6.3 show some increase in isolation chamber steam flow before or around 200 seconds. What is the cause of this increase? Reference DCD, Revision 2, Figures 6.3-12a, 6.3-20a, 6.3-28a, and 6.3-36a.
6.3-75	Klein V	Break flow for GDCS vs BDL breaks	The break areas for the gravity driven cooling system break and the bottom drain line break (BDL) are relatively comparable (0.004561m^3 vs 0.004052m^3 respectively). What is critical flow for these breaks? Are these breaks at critical flow? At what pressure difference would these breaks be at critical flow? The break flows in DCD, Tier 2, Revision 2, Figures 6.3-27b (BDL break) and 6.3-35b (gravity driven cooling system (GDCS) Line Break) show the GDCS line break having a substantially higher (more than 2 times) break flow for the first 200 seconds. Provide an explanation for the difference in break flows. Explain the dip in GDCS break flow from about 8 to 28 seconds in Figure 6.3-35b.
7.7-6	Li H Carte N	Verify consistency between Chapter 7 and assumptions in the safety analysis in Chapter 15	(A) DC, Tier 2, Revision 2, Section 15.3.7.2, “Sequence of Events and System Operation for the Control Rod Withdrawal Error During Refueling event”, states that “when the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the REFUEL position, only one control rod can be withdrawn when the RC&IS [rod control and information system] SINGLE/GANG rod selection status is in the SINGLE rod selection mode. When the RC&IS SINGLE/GANG rod selection status is in the GANG rod selection mode, only one control rod pair with the same HCU [hydraulic control unit] may be withdrawn. Any attempt to withdraw an additional rod results in a rod block by the RC&IS interlock." DCD Tier 2, Revision 2, Section 7.7.2.2.7.4, "Rod Block Function," provides a list of the conditions which result in a rod block signal to the RC&IS. However, the rod block signals discussed in DCD Section 15.3.7.2 are missing from this list. Please revise DCD Tier 2, Section 7.7.2.2.7.4 to include these rod block interlocks and verify that all other rod block interlocks assumed in the analyses provided in DCD Tier 2 Chapter 15 are listed accordingly. Also, provide ITAAC in DCD Tier 1, Table 2.2.1-1, "ITAAC For Rod Control and Information System," for all rod block interlocks assumed in the analyses provided in DCD Tier 2 Chapter 15 analyses.

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			(B) DCD, Tier 2, Revision 2, Section 15.2.1.1, "Loss of Feedwater Heating," states that the Feedwater Control System (FWCS) Logic is provided in subsection 7.7.3, and includes logic to mitigate the effects of a loss of feedwater heating capability. However in DCD, Tier 2, Revision 2, Section 7.7.3, there is no such description of FWCS logic, and no logic diagram is provided. Therefore, in this respect, Chapter 7 & 15 are inconsistent. Please update DCD, Tier 2, Section 7.7.3 accordingly. (C) DCD, Tier 2, Revision 2, Section 7.2.1.2.4.2, “Initiating Circuits,” lists turbine stop valve (TCV) fast closure as “any one or more of the conditions listed below” that the “RPS logic initiates a reactor scram.” DCD, Tier 2, Revision 2, Section 15.2.2, “Increase in Reactor Pressure” identifies several operational occurrences causing this and the resulting overall system responses. It states that if “the control system verifies that bypass capacity is adequate, the system will activate the SCRRI to reduce the power to 60 percent “. In other situations it explains, “an increase in system pressure and reactor shutdown (will happen) if the available turbine steam bypass capacity is insufficient.” Therefore a scram is only imminent, after TCV fast closure, if there is insufficient steam bypass capability. Please address this apparent inconsistency between Chapter 7 and 15 and verify that a reactor scram following TCV fast closure would occur only if there is insufficient steam bypass capability. (D) Inconsistencies have been previously identified by the staff between DCD Chapters 7 and 16. In light of these inconsistencies and the inconsistencies identified above between Chapters 7 and 15, please inform the staff of corrective actions that have been taken to ensure consistency within the DCD.
16.2-111	Dehmel JC	The title of TS Section 5.5.6 is not consistent with its basis document	The title of DCD, Revision 2, Chapter 16, Section 5.5.6 of the ESBWR Technical Specifications (TS) is not consistent with NUREG-1434 on which the ESBWR TS are based. Please address this inconsistency.

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20.0-7	Thomas G	Explain in detail, the systems design, operation and operator actions during transients, and demonstrate that the RPV level system is robust	Inadequate Core Cooling (TMI-2 Action Item II.F.2) In view of the importance of reactor pressure vessel (RPV) level instrumentation for safety in BWRs, and the staff concerns reflected in GL-84-23, GL-92-04 and Bulletin 93-03, combined with the fact that ESBWR is a passive, natural circulation plant, confirmation is needed regarding the adequacy of the differential pressure (dp) method for the RPV level measurement. Please explain in detail, the systems design, operation and operator actions during transients, and demonstrate that the RPV level system is robust. For example, the vertical drop in the drywell for the ESBWR RPV water level reference leg instrument lines from the condensing chamber to the drywell wall is an important design detail that should be provided for review.
21.6-95	Klein V	Changes to TRACG04 versions 42 to 45	During the NRC Staff audit of TRACG as applied to ESBWR loss of coolant accident on December 11-15, and 19-20, the audit team did not locate any documented changes from TRACG04 versions 42 to 45. Verify that there were no substantial changes to TRACG04 for these versions.
21.6-96	Klein V	Concerns about using the PC version of TRACG04 which has not yet been fully qualified instead of the Alpha VMS version of the code	During the NRC Staff audit of TRACG as applied to ESBWR loss of coolant accident (LOCA) on December 11-15, and 19-20, GE stated that GE is using the PC version of TRACG04 for ESBWR LOCA analyses. The audit team viewed a document on the comparison of TRACG04A (Alpha VMS version) to the TRACG04P (PC) version. ("Comparison of TRACG Results for ESBWR ECCS & CONT Cases - PC versus ALPHA versions," DRF 0000-0054-3548 Section 0000-0055-6820, July 19, 2006) Please address the following: A. State what version of TRACG04 (A or P) is being used for all ESBWR analyses using TRACG in DCD Chapters 4, 6 and 15. B. The TRACG04A and P comparison that the staff viewed during the audit was for the limiting breaks in DCD Rev. 1, show the differences between TRACG04A and P for the limiting breaks in the most recent version of the DCD using the updated nodalizations. C. The comparison between TRACG04A and P shows that TRACG04P predicts a long term drywell containment pressure lower by roughly 20kPa (or 3psi). For the DCD, Rev. 1 analyses the peak pressure was reached in the short term. For that calculation you stated

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			that the long term differences were not important for the peak pressure calculation. Rev. 2 of the DCD shows that peak pressure is reached in the long-term. Address the possible non-conservatism between TRACG04A and P for the long term peak pressure analysis in the latest revision of the DCD. D. In your comparison between TRACG04A and P, you state that the reason for the difference in wetwell and drywell pressures was due to roundoff errors in non-condensible gas concentrations. The NRC staff is concerned that roundoff errors can have a substantial (roughly 7%) effect on calculated peak pressures. Address the concern that the TRACG04 and/or the ESBWR LOCA model may be hyper-sensitive to non-condensible gas concentrations.
21.6-97	Klein V	Submit the most recent version of the TRACG04 User's Manual	Step 5 of the CSAU methodology requires that complete code documentation be provided for the frozen code version. The staff currently has NEDC-32956P, Rev. 0, "TRACG04A,P User's Manual," DRAFT July 2005. Please provide the updated version of this document.
21.6-98	Klein V	Address the remaining confirmatory items for the NRC's SER on TRACG for LOCA Analyses for ESBWR	The staff noted in its acceptance review of ESBWR (Reference 1) that GE did not address all of the confirmatory items that were to be performed at the Design Certification stage as stated in the Staff's SER on TRACG for ESBWR loss of coolant accident (LOCA) analyses (Reference 2). In response to the staff's acceptance review of ESBWR, GE submitted some information (Reference 3) to address the confirmatory items in Reference 2, but this information is still incomplete. Please address the following confirmatory items: 2. Submit the long-term core cooling analyses. 13. Analyze standard problems and submit to the NRC. 14. Provide all nodalization changes including diagrams since the approval of TRACG for ESBWR LOCA Analyses in Reference 2, include most recent changes incorporated into Rev. 2 of the DCD; Explain the statement in Reference 3 that a "Total of 5 chimneys to calculate the minimum water level." In the TRACG input decks submitted to the staff and in Figures 6.2-6 and 6.2-7, the core/chimney section is divided into only 3 rings. 19. GE needs to submit additional information on the passive containment cooling system (PCCS) vent system demonstrating that it will perform as expected.

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			20. Describe all design changes since the approval of TRACG for ESBWR LOCA Analyses in Reference 2 and demonstrate that the staff's conclusions would not be altered as a result of these changes. References: 1. Letter to S.A. Hucik (GE) from W.D. Beckner (NRC), "Results of Acceptance Review for ESBWR Design Certification Application (TAC No. MC8168)," September 23, 2005 2. Letter to L.M. Quintana (GE) from W.D. Beckner (NRC), "Reissuance of Safety Evaluation Report Regarding the Application of General Electric Nuclear Energy's TRACG Code to ESBWR Loss-of-Coolant Accident (LOCA) Analyses (TAC NOS. MB6279, MB6280, MB6281, MB6282, MB6283, MB6801 and MB7255)," October 28, 2004 3. Letter from D.H. Hinds (GE) to NRC, MFN 05-096, "Summary of September 9, 2005 NRC/GE Conference Call on TRACG LOCA SER Confirmatory Items," September 20, 2005
21.6-99	Klein V	Justify the use of the boron mixing model	During the NRC staff's audit of TRACG as applied to ESBWR LOCA, the staff reviewed a document (Reference 1) that contained a GE internal review of the TRACG qualification, as part of GE QA processes. This document stated that the TRACG application statement should document that the boron mixing model is not qualified and its use is not recommended. GE confirmed that this was added to the application statement in the TRACG04 User's Manual. Please explain this statement and GE's subsequent use of the boron mixing model in ESBWR ATWS applications (Reference 2). References: 1. "TRACG04A Qualification Design Review Closure Items," DRF 0000-0041-0817 2. NEDE-33083P Supplement 2, "TRACG Application for ESBWR Anticipated Transient Without Scram Analyses," January 2006

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21.6-100	Klein V	Provide more details on the CHAN leakage model used in the ESBWR ATWS analyses	Please answer the following questions regarding the CHAN leakage model: A. Are you using the "GE Design Leakage Flow correlations" derived from Reference 1 to calculate leakage flow in the ESBWR anticipated transients without scram (ATWS) calculations? Since this correlation was derived for loss of coolant accident (LOCA) conditions, is it applicable for high pressure conditions, such as those seen in an ESBWR ATWS event? Provide the correlation's applicability range. B. The TRACG04 ESBWR ATWS input decks indicate that GE may be overlaying the default values of coefficients for the GE Design Leakage Flow correlations via specification of CWF and CWB. If so, justify the selection of these coefficients. C. The CHAN leakage model described in Section 7.5.1 in Reference 2 is based on a driving pressure for each of the leakage paths. Provide a discussion on how TRACG04 selects the reference pressures where the leakage flow is calculated. Include details such as the cells used for calculating these pressures. References: 1. 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. 2. NEDE-32176P, Revision 3, "TRACG Model Description," April 20, 2006

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 85 RELATED TO ESBWR DESIGN CERTIFICATION APPLICATION

ORIGINATOR: S. Williams

SECRETARY: C. Nagel

DATE: January 23, 2007

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