21. TESTING AND COMPUTER CODE EVALUATION

21.1 Introduction

The General Electric economic simplified boiling-water reactor (ESBWR) design is the first natural circulation cooled boiling-water reactor (BWR) reviewed by the U.S. Nuclear Regulatory Commission (NRC). The design incorporates many unique features, including a very large chimney arrangement above the core and a passive gravity-driven emergency core cooling system (ECCS). The features of the design necessitated an extensive review of the testing program and accident and transient analysis computer codes to establish their applicability to the ESBWR. However, since core uncovery is not expected for any postulated break in the piping attached to the reactor pressure vessel (RPV), it was not necessary for the staff to review models and correlations for estimating fuel heatup.

The test and analysis program description (TAPD) in NEDC-33079P, "ESBWR Test and Analysis Program Description," Class III, Revision 1, November 2005, (NEDC-33079P) provides an integrated plan to address the experimental and analytical work needed to analyze ESBWR performance for normal operations, transients, design-basis accidents (DBAs), stability, and anticipated transient without scram (ATWS) conditions in support of ESBWR design certification. A major product of all these activities is the assessed TRACG code for ESBWR analysis. The preapplication review of the ESBWR focused on the review of the TRACG code for loss-of-coolant accident (LOCA) and containment analysis; the staff safety evaluation documents the results of this review in Letter from W. D. Beckner (NRC) to L. M. Quintana (GENE), "Re-Issuance of Safety Evaluation Report Regarding the Application of General Electric Nuclear Energy's TRACG Code to ESBWR Loss-of-Coolant (LOCA) Analyses," October 28, 2004. The preapplicaton review did not include an evaluation of the test data and TRACG qualification for operational transients, ATWS, and stability. This chapter of the safety evaluation report (SER) discusses the applicability of the GE-Hitachi Nuclear Energy Americas LLC (GEH) testing program and scaling analysis to the updated ESBWR design submitted in the design control document (DCD) regarding LOCA performance, in addition to summarizing the key findings of the preapplication review.

As required by Title 10, Section 52.47(b)(2)(c)(2), of the *Code of Federal Regulations* (10 CFR 52.47(b)(2)(c)(2)), GEH used experimental data from a number of basic and separate effects tests with generic applications to operating BWRs and the ESBWR, full-size component tests and integral systems tests performed specifically for the simplified boiling-water reactor (SBWR) and ESBWR, and BWR plant operation to qualify the TRACG code for the ESBWR LOCA analyses. Section 21.5.3 of this report summarizes the test data (excluding the basic tests) used to qualify the TRACG code initially for the SBWR and now for the ESBWR LOCA applications. The facilities described were designed and scaled based on the SBWR design. The NRC staff reviewed the facilities for their applicability to the ESBWR design. The staff conclusions regarding applicability are based on a review of the test objectives, test descriptions, phenomena represented, and adequacy of the GEH scaling analysis, as discussed in Section 21.5.3 of this report. This assessment references the SBWR as well as the ESBWR design since the facilities were originally designed relative to the SBWR.

21.2 Limitations and Restrictions

Many of the safety systems of the ESBWR design represent new concepts in plant safety system design. As a result, an extensive scaled facility testing program was developed for the predecessor concept, the SBWR design. While the SBWR was considerably smaller than the ESBWR, the two incorporate many system similarities. As noted in the introduction to this chapter, the ESBWR design does not experience core uncovery for any postulated LOCA. The staff has reviewed the accident analysis computer codes with this basis. Should there be any changes that result in core uncovery, the conclusions regarding the acceptability of the computer codes will be invalid.

GEH scaled and designed the systems test facilities in such a way that little data were obtained regarding multidimensional phenomena. In addition, the GEH scaling analysis was based on the lumped-parameter technique. As such, the tests did not provide sufficient data to qualify the TRACG code for multidimensional spatial variations. Without multidimensional capability, the TRACG code is unable to accurately predict drywell mixing, noncondensible gas stratification, or buoyancy/natural circulation inside the containment. As a consequence of this limitation, the TRACG code will employ conservatively bounding ESBWR containment models. (For further discussion, see the staff evaluation presented in Section 21.5.3 of this report.)

21.3 Overview of General Electric Testing Programs

In DCD, Tier 2, Section 1.5.3, GEH described the ESBWR test program. NEDC-33079P provides the ESBWR TAPD, which includes a detailed justification for the adequacy of the test database for application to the safety analysis. The phenomena identification and ranking table (PIRT) discussed in Section 2 of NEDC-33079P identifies specific governing phenomena, of which a significant fraction was concluded to be "important" in predicting ESBWR transient and LOCA performance. Most of these phenomena are common to those for operating BWRs. TRACG has been extensively qualified against separate effects tests, component performance tests, integral systems tests, and plant operating data listed in NEDC-33079P. The TRACG qualification report NEDE-32177P, "TRACG Qualifications," Class III, Revision 2, January 2000, (NEDE-32177P) documents this "base" qualification. This section describes specific SBWR/ESBWR-related tests and test facilities beyond the previous qualification database.

GEH stated that, while all SBWR/ESBWR features are extrapolations from current and previous designs, two features (specifically, the passive containment cooling system (PCCS) and the gravity-driven cooling system (GDCS)) represent the two most challenging extrapolations. Therefore, it was decided that, for these two cases, it was necessary to obtain additional test data that could be used to demonstrate the capabilities of TRACG to successfully predict SBWR/ESBWR performance over a range of conditions and scales. "Blind" pretest analyses of selected test conditions using only the internal correlations of TRACG were performed before the start of testing. "Blind" indicates that the analyst had no information on the results of the experiments. No "tuning" of the TRACG inputs was performed, and no modifications to the coding were anticipated as a result of these tests. A number of "double blind" pretest analyses were also performed for certification data experiments. "Double blind" indicates that the analyst had no information on either the results or the exact initial conditions of the experiments. These predictions were based on the as-designed facility configurations.

For the PCCS, the steady-state heat exchanger performance was predicted in full-vertical-scale 3-tube (GIRAFFE), 20-tube (PANDA), and prototypical 496-tube (PANTHERS) configurations, over the range of steam and noncondensible conditions expected for the SBWR. This process addressed scale and geometry differences between the basic phenomena tests performed in single tubes and larger scales, including prototype conditions. Transient performance was similarly investigated at two different scales in both GIRAFFE and PANDA.

TRACG GDCS performance predictions were performed against the GIST and GIRAFFE/SIT test series. Pretest predictions have also been performed for the PANTHERS and PANDA steady-state tests.

GEH further stated that, in some cases, the ESBWR DCD did not delineate the detailed design of specific ESBWR plant equipment; in some instances, only the design requirements of the equipment were given. When this is the case, a requirement for testing of specific hardware is not required prior to design certification. However, testing of plant-specific hardware will be done before or during startup testing of the plant as part of the completion of the ITAAC provided in DCD Tier 1 and the initial test program described in DCD Tier 2, Section 14.2. For example, the plant startup test program will include the overall testing of the heat rejection capability of the isolation condenser system (ICS). Plant-specific startup tests will be conducted to confirm that each ICS meets the performance requirements prior to commercial operation as specified in Section 14.2.8.2.34 of the ESBWR DCD. Full-scale tests of an ICS module in the PANTHERS test facility, as well as experience with condensing heat exchangers in many industries, offer a high degree of confidence that the requirements will be met.

21.3.1 Major ESBWR Unique Test Programs

As indicated in the DCD, the vast majority of data supporting the ESBWR design were generated using the design of the previous BWR product lines. ESBWR-unique certification and confirmatory tests applicable to its design, as described in the DCD, are provided below.

The staff evaluation of ESBWR test programs in Section 21.5.3 of this report fully discusses testing issues unique to the ESBWR.

21.3.1.1 MIT/UCB Single Tube Condensation Test Program

Early in the SBWR program, researchers identified that information was needed to determine a heat transfer correlation for steam condensation in tubes in the presence of noncondensible gases. A test program was conducted to secure this information. This test program is documented in NEDC-32301, MIT and UCB Separate Effects Tests for PCCS Tube Geometry, “Single Tube Condensation Test Program,” March 1994 (NEDC-32301).

The Single Tube Condensation Test Program was conducted to investigate steam condensation inside tubes in the presence of noncondensibles. The work was independently conducted at the University of California at Berkeley (UCB) and the Massachusetts Institute of Technology (MIT). The work was initiated to obtain a database and a correlation for heat transfer in conditions similar to those that would occur in the SBWR/ESBWR PCCS tubes during a DBA LOCA. UCB researchers utilized three separate experimental configurations, and MIT researchers utilized one configuration. The researchers ran tests with pure steam, steam/air, and steam/helium mixtures with representative and bounding flow rates and noncondensible mass fractions. The results demonstrated that the system behaved as expected for the tests with conditions similar to the ESBWR design. The results of the tests at UCB became the basis for the condensation heat transfer correlation used in the TRACG computer code.

21.3.1.2 GIST Test Program

GIST is an experimental program conducted by GEH to demonstrate the GDCS concept and to collect data to qualify the TRACG computer code for ESBWR applications. Simulations were conducted of DBA LOCAs representing a main steam line break (MSLB), bottom drainline break (BDLB), GDCS line break, and a no break scenario (e.g., a loss of feedwater).

Test data, documented in GEFR-00850, “Simplified BWR Program Gravity-Driven Cooling System (GDCS) Integrated Systems Test,” October 1989 (GEFR-00850), have been used in the qualification of TRACG to the ESBWR. Tests were completed in 1988 and documented by GEH in 1989. GIST data is used to validate certain features of the TRACG code.

21.3.1.3 GIRAFFE Test Program

The GIRAFFE Test Program, documented in NEDC-32606P,“SBWR Testing Summary Report,” Class III, November 1996, (NEDC-32606P) is an experimental program conducted by the Toshiba Corporation to investigate the thermal-hydraulic aspects of the PCCS. Fundamental steady-state tests on condensation phenomena in the PCCS tubes were conducted. Simulations were run of DBA LOCAs, and specifically, the MSLB. GIRAFFE data have been used to substantiate PANDA and PANTHERS data at a different scale and to support validation of certain features of TRACG. Also, two additional series of tests have been conducted in the GIRAFFE facility. The first (GIRAFFE/Helium) test demonstrated the operation of the PCCS in the presence of lighter-than-steam noncondensible gas; the second (GIRAFFE/SIT) test provided additional information regarding potential system interaction effects in the late blowdown/early GDCS period.

21.3.1.4 PANDA Test Program

The PANDA Test Program, documented in NEDC-32606P, is an experimental program run by the Paul Scherrer Institute (PSI) in Switzerland. PANDA is a full-vertical-scale, 1/25-volume-scale model of the SBWR system designed to model the thermal-hydraulic performance and post-LOCA decay heat removal by the PCCS. Both steady-state and transient performance simulations have been conducted. Testing at the same thermal-hydraulic conditions as previously tested in GIRAFFE and PANTHERS allows scale-specific effects to be quantified. Blind pretest analyses using TRACG were submitted to the NRC before the start of testing. PANDA data is used to validate certain features of the TRACG code.

21.3.1.5 PANTHERS Test Program

The PANTHERS Test Program, documented in NEDC-32606P, is an experimental program performed by Ente Nazionale per l’Energia Electtrica at Societa Informazioni Esperienze Termoidrauliche (SIET) in Italy, with the dual purpose of providing data for TRACG qualification and demonstration testing of the prototype PCCS and ICS heat exchangers. Steam and noncondensibles were supplied to prototype heat exchangers over the complete range of SBWR conditions to demonstrate the capability of the equipment to handle post-LOCA heat removal. Testing was performed at the same thermal-hydraulic conditions as those used in the GIRAFFE and PANDA testing. Blind pretest analyses of selected test conditions using TRACG were submitted to the NRC before the start of testing. PANTHERS data is used to validate certain features of the TRACG code.

In addition to thermal-hydraulic testing, an objective of PANTHERS was to demonstrate the structural adequacy of the heat exchangers to exceed the SBWR/ESBWR expected lifetime requirement. GEH stated that this was accomplished by pre- and post-test nondestructive examination, following cycling of the equipment in excess of requirements.

21.3.1.6 Additional PANDA Tests

A supplementary test program, documented in NEDC-33081P, "ESBWR Test Report," Class III, Revision 1, May 2005, was also performed in the PANDA test facility to evaluate an earlier ESBWR configuration with the GDCS pool connected to the wetwell gas space rather than the drywell. These tests confirmed the expected increased margin to the containment design pressure for this ESBWR configuration. This series of tests also included injection of helium, providing data on PCCS performance with light noncondensible gases at an additional scale.

21.3.2 Scaling of Tests

GEH discussed the effects of scaling on the major SBWR and ESBWR tests in NEDC-32288P, "Scaling of the SBWR Related Tests," Class III, Revision 1, October 1995; NEDC-33082P, Revision 1, "ESBWR Scaling Report," Class III, January 2006, and MFN 06-225, "Response to NRC Request for Additional Information Letter No. 4 Related to ESBWR Design Certification Application – ESBWR Scaling Analysis – RAI Number 6.3-1," July 18, 2006. These reports assess the features and behavior of the SBWR and ESBWR during challenging events. The analysis included the general (top-down approach) scaling considerations, the scaling of specific (bottom-up approach) phenomena, and the scaling approach for the ESBWR-specific tests.

The staff evaluation of GEH scaling methodology in Section 21.5.3 of this report provides a full discussion of scaling issues.

21.4 Overview of NRC Activities on the Test Programs

Thermal-hydraulic test programs unique to the ESBWR design were used to support qualification of analytical codes used in the design and licensing of the ESBWR nuclear power plant. These tests were performed as a part of the prior SBWR plant design. The NRC performed quality inspections of these test programs as a part of the oversight activities needed to prepare the SBWR design for a licensing submittal.

These activities included observing selected tests at some of the SBWR test facilities and auditing the applicant's performance of a broad range of issues related to the following:

- test facility design, instrumentation, and scaling
- test data and analyses
- quality assurance (QA)

During the inspections, procedural defects were noted and corrected, and in the end, the programs were determined to meet appropriate quality requirements. On the basis of its observations of these tests, as documented in test observation reports, the staff concluded that the applicant performed the design certification test programs in a competent, professional manner, with due consideration given to meeting the test specifications and acceptance criteria.

The staff believed that the test programs provided useful data for evaluating the ESBWR passive safety system performance; however, the staff did perform a detailed review of the test results to reach a final judgment on the adequacy of the vendor's test programs. As discussed in Section 21.5 of this report, the staff, based on its evaluation, concluded that the SBWR testing is also applicable to ESBWR design certification. Section 21.6.2.6 of this report provides a comprehensive summary of the QA inspections. The following sections summarize the course of the quality inspection activities for the test programs.

21.4.1 GIRAFFE Test Programs

The Toshiba Corporation performed three separate sets of tests at its Nuclear Engineering Laboratory (NEL) in Kawasaki City, Japan, in support of the SBWR. The gravity-driven integral full-height test for passive heat removal (GIRAFFE) test facility included all the major components of the SBWR design and had the capability to perform both steady-state, component performance, and transient system response testing.

The first series of GIRAFFE tests (hereafter designated GIRAFFE Phase 1) were performed as development tests to confirm the operational feasibility of the SBWR concept and did not include the level of QA expected of a design-basis test. For this reason, only limited segments of the database were used in support of the SBWR and only for comparison to the steady-state passive containment cooling (PCC) performance tests of PANDA and PANTHERS.

The GIRAFFE/Helium and GIRAFFE/SIT series were transient system performance tests run to design-basis QA standards. These tests investigated the effects of lighter-than-steam noncondensible gases on PCCS performance and potential systems behavior (e.g., isolation condenser (IC) operation during LOCA), respectively.

The NRC staff traveled to the Toshiba NEL in Kawasaki (about 15 miles south of Tokyo) for further discussions about the GIRAFFE SIT and "H"-series tests and to observe the performance of test GS-2 as documented in MNF 276-95, "GIRAFFE/SIT Trip Report dated 10/27/95," November 7, 1995. The test was nominally a repeat of GS-1 (i.e., a double-ended guillotine break of a GDCS injection line), but with actuation of the PCCS and ICS. However, the discussions also included coverage of the preliminary results of a "shakedown" run of GS-2, which had been performed earlier.

The pretest procedures in GIRAFFE are relatively complex because of the necessity to initialize a test "on the fly." These complexities are increased when the ICS is used. GS-2 was the first test ever performed in GIRAFFE in which the ICS was brought on line at the test initiation pressure of about 1 megapascal (MPa). Toshiba determined that it would be difficult to allow the ICS to operate during test initiation, with the condensate returning directly to the RPV, while maintaining pressure in the RPV. Thus, the ICS return to the RPV was valved off, and condensate was allowed to collect in the heat exchanger. If the accumulated condensate was allowed to flow to the RPV when the ICS was brought on line, it would distort the previously established RPV water level at the start of the test. Thus, the IC was drained outside the facility immediately before test initiation to allow proper loop conditions to be established. This procedure had not been tested before performance of GS-2. Therefore, Toshiba performed a shakedown run of that test to determine whether the ICS startup procedures accomplished the desired result. In addition, data were collected as though the shakedown run was an actual matrix test.

Post test evaluation of the data demonstrated, to Toshiba's satisfaction, that the ICS startup procedure was successful, and the data were presented at the NEL as a "preview" of what would likely be seen when the test was performed "officially" later that day. The results of both tests are discussed further below.

The final activities at the NEL were to review the shakedown run of GS-2 and the observations of the official test run. Toshiba had plotted some of the key data from the shakedown run for comparison to both GS-1 and TRACG pretest analyses. To some extent, the response of GS-1 and GS-2 were similar, especially near the start of the test. The minimum water level in the RPV was not as low as in GS-1. This was due, in part, to a higher starting level, the value for which was determined from an analysis of the event. The ICS return valve opens before actuation of the automatic depressurization system (ADS) so that any accumulated water in the IC tubes and outlet plenum enters the RPV. This adds inventory and also helps depressurize the RPV. As a result, the predicted water level in the RPV when the pressure reaches about 1 MPa, which is used to determine that parameter in GIRAFFE, is somewhat higher than if the ICS is not employed.

Other trends in the two tests were quite similar. The drywell and wetwell pressure curves in GS-1 and GS-2 were of the same general shape, with condensation in the drywell occurring because of the injection of GDCS water through the broken line. As a result, the wetwell pressure stayed higher than the drywell pressure, again causing numerous actuations of the vacuum breaker (VB); however, since the PCCS operates only when the drywell pressure is greater than the wetwell pressure, that system did not play a substantial role until very late in the transient. The peak containment pressure in GS-2 was about 10 percent lower than in GS-1, partly as a result of IC heat removal. The drywell and wetwell pressures began to increase shortly after the cessation of GDCS flow to the drywell (at 1 hour). In GS-1, after GDCS injection to the RPV had ended, steam production resumed in the RPV, and the venting of that steam through the ADS to the drywell brought the drywell pressure above that of the wetwell. Since the PCCS was shut off in GS-1, there was no energy removal to reduce the drywell pressure, and, near the end of the test, the drywell pressure exceeded the wetwell pressure sufficiently to open the LOCA vents. In GS-2, the ICS and PCCS were both available to remove energy once steam production resumed, and the PCCS operation prevented the drywell from reaching a pressure sufficiently greater than that of the wetwell to open the LOCA vents. In this test, therefore, the PCCS performed its function in limiting both overall drywell pressure (up to 2 hours post-LOCA) and the pressure difference between the drywell and the wetwell. There were no apparent detrimental systems interactions, and safety-related injection and heat removal systems operated as designed.

Observation of the official run of the GS-2 test began shortly before test initiation. All loop manipulations required to "fine tune" the facility before test initiation can be done from the control room using remote manual actuation of facility components. A small control room staff was required to accomplish those tasks. The written procedures were followed closely, and steps were noted on a test log/checklist, which was signed by the test engineer. Toshiba staff operated professionally, and there appeared to be appropriate consideration of testing QA throughout the observed portion of the test initialization process and during the performance of the experiment. The staff was able to track key parameters, such as wetwell and drywell pressures, GDCS flow, selected temperatures, and water levels through control room digital displays or analog chart recorders. The GDCS initiation time, GDCS flow rate, RPV water level, and approximate pressure-time response of the wetwell and drywell agreed very closely with the results from the previous day's shakedown run. Therefore, the two tests provided an indication of data repeatability, which was also valuable to the staff's assessment of the test program.

The GIRAFFE "H"-series and GIRAFFE/SIT tests comprised well run test programs and were conducted with appropriate attention to QA concerns. Section 21.5.3 of this report more fully discusses the staff's concerns associated with some issues, such as scaling and test control (e.g., microheater power). However, the data provided by these test programs were useful for code validation as part of the SBWR/ESBWR design certification effort.

21.4.2 PANTHERS Test Programs

As part of the SBWR design process, SIET and the European Nuclear Energy Association (ENEA) tested full-size prototype heat exchangers for the PCCS and ICS at the performance analysis and testing of heat removal system (PANTHERS) test facility in Piacenza, Italy. Ansaldo Spa designed and built the prototype PCCS and IC heat exchangers.

A readiness assessment was conducted for the PANTHERS/PCCS test program at SIET and the staff reviewed the initial readiness assessment report. The purpose of the assessment was to ensure the technical adequacy of the facility and personnel to conduct the planned tests in accordance with test requirements. A specific goal was to ensure that all preparations were either complete or proceeding so that the test could be initiated with a high degree of confidence that quality results would be obtained. The assessment team concluded that personnel assigned to perform the tests were technically capable of conducting the test according to the requirements. Procedures and associated QA practices were in place and adequate to control the work.

The staff visited the SIET facility in Piacenza, Italy, to observe testing in the PANTHERS-PCCS facility for the GEH SBWR design as documented in MNF 170-94, "Summary of the visit on October 16, 1994, at the Societa Informationi Esperienze Termoidrauliche (SIET) Performance Analysis and Testing of Heat Removal System (PANTHERS) Test Facility for the SBWR Design," December 21, 1994. Major observations from the visit are discussed below.

Testing in PANTHERS provided considerable data on PCCS heat exchanger performance. The testing was under the supervision of both GEH and ENEA (which was a partner in ownership of SIET Laboratories) and was performed by a SIET team different from the one operating the SPES-2 facility. It was difficult to generalize on the basis of a single test, but the test operations crew demonstrated the same sort of competence and professionalism in PANTHERS testing as was noted previously for the operation of the SPES-2 facility.

The specific test observed by the staff involved measurement of the heat transfer capability of the PCCS unit with a steam-air mixture. In addition to degradation of heat transfer by the noncondensible gas, the water level in the PCCS surrounding the heat exchanger was lowered very gradually to determine the effect of that parameter on heat transfer performance. Observers noted very little effect of the lowered water level until a significant fraction (less than 50 percent) of the tube surface was uncovered. The staff believed that the observation of these activities was valuable in preparing for future observation of ICS testing. Staff had some concerns regarding the ICS structural integrity and design, particularly the leakage in the ICS during testing at the PANTHER-IC facility. Staff considered this an ICS structural integrity issue that needed to be resolved for the ESBWR design certification.

Section 21.5.3 of this report discusses the staff's concerns associated with some issues, such as the ICS structural integrity issue, and GEH's plan to resolve them.

21.4.3 PANDA Test Programs

Passive Nachwarmeabfuehr-und DrueckAbbau Testanlage (passive decay heat removal and depressurization test facility) (PANDA) testing was performed as a joint effort between GEH and the Paul Scherrer Instutut (PSI) in Wuerenlingen, Switzerland. The PANDA facility included all of the major components of the SBWR design and had the capability to perform both steady-state component performance and transient system response testing.

The PANDA S-series tests were steady-state performance tests of the PCC and IC heat exchangers to identify any scale effects on PCC heat exchanger performance. The PANDA M-series tests were integral systems transient performance tests to demonstrate startup and long-term operation of the PCCS and to investigate potential systems interaction effects.

Test readiness review of the PANDA facility and test program was performed. The staff attended the review as observers. The purpose of this assessment was to ensure the technical adequacy of the facility and personnel to conduct the PANDA tests in accordance with the test requirements. The assessment was divided into horizontal and vertical reviews. The horizontal review consisted of determining the overall readiness of the facility, its personnel, and documentation. The vertical review consisted of a more detailed examination of a part of the facility (e.g., a single instrument line, data calculation) to verify the technical adequacy and correctness of the work. This review was held early in the program development to ensure that adequate time was available to address any potential deficiencies. Section 21.5.3 of this report discusses the issues regarding the validity of the test data and the nonprototypical features of the model.

21.5 Evaluation of Vendor (GEH) Testing Programs

21.5.1 Regulatory Criteria

The following is required by 10 CFR 50.43(e) as referenced by 10 CFR 52.47(b)(2)(c)(2):

- The performance of each safety feature of the design has been demonstrated through analysis, appropriate test programs, experience, or a combination thereof.

- Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof.

- Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analysis over a sufficient range of operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

21.5.2 Summary of Technical Information in the Application

Section 21.3 of this report provides an overview of the vendor (GEH) testing program.

21.5.3 Staff Evaluation

21.5.3.1 Full-Size Component Tests

MIT/UCB Single Tube Condensation Test Program

The Single Tube Condensation Test Program was conducted to investigate steam condensation inside tubes in the presence of noncondensibles. The work was independently conducted at the UCB and MIT (NEDC-32301). The work was initiated to obtain a database and correlation for heat transfer in conditions similar to those that would occur in the SBWR/ESBWR PCCS tubes during a DBA LOCA. Researchers utilized three separate experimental configurations at UCB, while researchers at MIT utilized one configuration. Tests were run with pure steam, steam-air, and steam-helium mixtures with representative and bounding flow rates and noncondensible mass fractions. The results demonstrated that the system behaved as expected for all tests, with either of the noncondensibles, for forced flow conditions similar to the ESBWR design. The results of the tests at UCB are the basis for the condensation heat transfer correlation used in the TRACG computer code.

Condensation on PCCS primary side is a function of the mass flow rate and noncondensible fraction. The TRACG correlation is based on UCB test data. The correlation is qualified with UCB and MIT single tube, approximately full-length tests. PANTHERS provided confirmatory qualification.

The staff concludes that the experimental programs conducted for TRACG qualification of PCCS tube-side heat transfer are adequate for the condensation-driven mode. GIRAFFE and PANDA tests have shown that long-term containment performance is not highly sensitive to this correlation because the venting of noncondensibles allows the PCCS heat removal rate to match the reactor decay heat for the long term. The staff, therefore, finds that the test data is sufficient for developing the condensation heat transfer correlation used in the TRACG code, and meets the requirements of 10 CFR 50.43(e).

PANTHERS/PCCS Tests

This program tested a full-size PCCS condenser for the SBWR. The test objectives were to (1) demonstrate that the prototype PCCS heat exchanger for the SBWR was capable of performing as designed with respect to heat rejection (component performance), (2) provide a sufficient database to confirm the adequacy of TRACG to predict the quasi-steady heat rejection performance of a prototype PCCS heat exchanger over a range of airflow rates (to simulate nitrogen in the SBWR containment), steamflow rates, operating pressures, and superheat conditions that span and bound the SBWR (and ESBWR) range, and (3) determine and quantify any differences in the effects of noncondensible gas buildup in the PCCS heat exchanger tubes between lighter-than-steam and heavier-than-steam gases (concept demonstration).

A full-size PCCS condenser of the SBWR consists of two identical modules, and each module consists of a top header, a number of vertical condenser tubes, and a bottom header. The PANTHERS/PCCS tests provided data for a full-size, two-module PCCS condenser submerged in a pool of water. Although the tests focused on the performance of a PCCS condenser for the SBWR, the data are applicable to a PCCS condenser in the ESBWR, which has the same condenser tube diameter, length, and pitch as the condenser tested in PANTHERS for the SBWR. The only difference is that the PCCS condenser in the ESBWR has about 35 percent more tubes than does the SBWR. As a result, an ESBWR PCCS condenser is expected to have a heat removal rate about 35 percent higher than that measured in the PANTHERS/PCCS condenser.

PANTHERS/PCCS testing was performed as a joint effort by GEH, Ansaldo Spa, ENEA, and SIET in Piacenza, Italy. The test facility consisted of a prototype PCCS unit originally designed to represent the SBWR, a steam supply, an air supply, and vent and condensate volumes sufficient to establish PCCS thermal-hydraulic performance. The heat exchanger was a prototype unit, built by Ansaldo Spa using prototype procedures and prototype materials. The PCCS pool had the appropriate water volume for a prototypical PCCS unit.

For the steady-state performance tests, the facility was purged with steam and placed in a condition where steam or an air-steam mixture was sent to the PCCS and the flows of the condensate and vented gases were measured. Once steady-state conditions were established, data were collected for a period of approximately 15 minutes. Ninety-seven steady-state tests were performed, including the steam-only tests, with either saturated or superheated steam. Test conditions covered the entire range of the PCCS inlet flow rates and pressures expected in the SBWR.

Transient tests were conducted by first establishing steady-state conditions and then either varying the water level in the PCCS pool or allowing the unit to fill up from an injection of noncondensible gases with the ventline closed off by a blind flange.

Investigators evaluated several phenomena, including the overall PCCS heat removal rate, pool-water-level effect on PCCS performance, mass flow rate into the PCCS, condensation inside the tubes with or without the presence of noncondensible gases, poolside heat transfer, parallel PCCS tube effects, and parallel PCCS modules effects.

Full-size component tests were conducted with the test parameters covering those expected in the SBWR and ESBWR (after a 35-percent increase in the PCCS heat removal rate as tested to account for the approximately 35-percent increase in the number of condenser tubes) during LOCAs.

Test results demonstrated that a prototype PCCS heat exchanger for the ESBWR is capable of performing as designed with respect to heat rejection and provided a sufficient database to confirm the adequacy of the TRACG code to predict the quasi-steady heat rejection performance of a prototype heat exchanger over a range of airflow rates (as a simulant for nitrogen in the containment), steamflow rates, operating pressures, and superheat conditions that cover the expected ranges of values of the parameters for the ESBWR.

A large number of the tests were conducted at a pressure higher than the expected containment pressure in the ESBWR during a LOCA, such as MSLB, GDLB, or a BDLB. Also, lower pressure data bracket the expected range of the containment pressure in the ESBWR.

Researchers measured temperature at the inside and outside walls of four condenser tubes, but did not measure the bulk gas temperature inside these tubes. The heat transfer coefficient inside a tube cannot be derived from the test data. There were no measurements for mass flow rate and noncondensible gas concentration at the inlet of a condenser tube where tube wall temperatures were measured. As a result, a correlation between the heat transfer coefficient and the fluid velocity could not be derived from the test data. The results of the single tube condensation test program performed at UCB were the basis for the condensation heat transfer correlation used in the TRACG code. The UCB test program is discussed in Sections 21.3.1.1 and 21.5.3.1 of this report.

Since the PCCS tested at the PANTHERS-PCCS facility is equivalent to a full-size PCCS condenser in the SBWR, no scaling analysis was necessary and the test data provided a global heat removal rate for a full-size condenser in the SBWR. The PANTHERS/PCCS data confirmed that a PCCS condenser in the SBWR is capable of a heat removal rate of 10 megawatts (MW) (or higher depending on the inlet conditions) as designed. For the ESBWR, the heat removal rate of a PCCS condenser is expected to be around 11.0 MW (with 35 percent more condenser tubes than the one tested at the PANTHERS-PCCS facility).

The PANTHERS/PCCS tests were not designed to provide local thermal-hydraulic parameters, such as the heat transfer coefficient, mass flow rate, and noncondensible gas concentration, inside a condenser tube. As discussed above, the UCB test program provided the necessary data to qualify TRACG for these phenomena.

In conclusion, the staff believes that the PANTHERS/PCCS test data cover a broad range of the SBWR and ESBWR parameters, including inlet pressure, total mass flow rate, and total noncondensible gas concentration to confirm the PCCS heat removal rate under various LOCA conditions. Therefore, the PANTHERS/PCCS data are acceptable as a valid database to qualify the TRACG code for the global heat removal rate of a PCCS condenser under the expected LOCA conditions in the ESBWR.

PANTHERS/ICS Tests

An ICS unit consists of two identical modules, with each module comprising a top header, a number of vertical condenser tubes, and a bottom header. The PANTHERS/ICS tests provide data for one full-size module (half) of the ICS condenser submerged in a pool of water. Note that an ICS in the ESBWR is identical to the ICS in the SBWR tested in the PANTHERS/ICS tests.

The test objectives were to (1) demonstrate that the prototype ICS heat exchanger is capable of performing as designed with respect to heat rejection, (2) provide a sufficient database to confirm the adequacy of TRACG to predict the quasi-steady heat rejection performance of a prototype ICS heat exchanger over a range of operating pressures that span and bound the ESBWR range, (3) demonstrate the startup of the ICS unit under anticipated transient conditions, and (4) demonstrate the capability of the ICS design to vent noncondensible gases and to resume condensation following venting.

PANTHERS/ICS testing was performed at SIET in Piacenza, Italy. The facility consisted of a prototype ICS module, a steam supply vessel simulating the SBWR reactor vessel, a vent volume, and associated piping and instrumentation sufficient to establish ICS thermal-hydraulic performance.

The ICS tested was one module of a full-scale, two-module vertical tube heat exchanger designed and built by Ansaldo Spa. Only one module was tested because of the high energy rejection rate of the ICS unit and inherent limitations of facility and steam supply size. The ICS was a prototype unit, built using prototypical procedures and prototypical materials. The SBWR has six modules (three heat exchanger units). The ICS was installed in a water pool having one-half the appropriate volume for one SBWR ICS assembly.

For the steady-state tests, the steam supply to the steam vessel was regulated such that the vessel pressure stabilized at the desired value. A constant water level was maintained in the pressure vessel by draining condensate back to the power plant. Data were acquired for a period of approximately 15 minutes. Then the steam supply was increased or decreased to gather data at a different operating pressure or testing was terminated. In all cases, flow into the ICS was natural circulation driven, as is the case for the SBWR/ESBWR.

As with the PCCS tests, transient tests were conducted by first establishing steady-state conditions and then either varying the water level in the ICS pool or allowing the unit to fill up from an injection of noncondensible gases. The gases were subsequently purged through ventlines located on both the lower and upper headers.

In terms of phenomena, investigators evaluated the ICS heat removal rate, pool-water-level effect on the ICS performance, mass flow rate into the ICS, and poolside heat transfer.

Full-size component tests were conducted with the test parameters covering those expected in the ESBWR during both normal and accident conditions. Since the ICS tested has one of the two identical modules of a full-size ICS, a scaling analysis was not necessary and the test data were directly applicable to an ICS in the ESBWR (which has twice the heat removal rate as the ICS tested at the PANTHERS-IC facility).

Researchers measured temperature at the inside and outside walls of eight condenser tubes, but did not measure the bulk gas temperature inside these tubes. As a result, the heat transfer coefficient inside the tubes could not be derived from the test data. There were no measurements of the mass flow rate and noncondensible gas concentration at the inlet of a condenser tube where tube wall temperatures were measured. As a result, a correlation between the heat transfer coefficient and the fluid velocity could not be derived from the test data. Because it was not a test objective, the staff finds this acceptable. As discussed above the UCB test program provided the necessary data to qualify TRACG for these phenomena.

Staff had some concerns regarding the ICS structural integrity and design, particularly the leakage in the ICS that occurred during testing at the PANTHERS-IC facility. This was considered an ICS structural integrity issue that needed to be resolved for the ESBWR design certification. GEH stated that the O-ring design had been changed to a Helicoflex self-energizing O-ring design that is more resilient to distortion. GEH further stated that closing of the condensate return valve will be controlled to limit the gradients associated with shutdown and cooldown of the ICS heat exchanger. However, Table 14.2-1 of DCD Tier 2, "Power Ascension Test Matrix," indicated that the ICS performance test will be conducted at a medium-power level, but not at a high-power level. Because one of the objectives of a power ascension test should be to demonstrate ICS structural integrity, the staff believes that an ICS performance test at high power would be of more value because the operating conditions at high power are expected to be more challenging to the structural integrity of ICS. The staff, therefore, requested in RAI 14.2-3 that the ICS performance test be conducted at high power, rather than at a medium-power level.

In response, GEH stated that the ascension test matrix (Table 14.2-1 of DCD Tier 2) proposed that the ICS be tested at medium (up to about 75-percent rated) power. Pressure and temperature affects the structural integrity of the ICS, not the reactor power level. When the reactor startup begins, the reactor is brought to the rated pressure and temperature at approximately 5-percent power, as stated in DCD Tier 2, Section 14.2.1.3. As the power level increases, the same rated pressure and temperature are maintained; therefore it is sufficient to conduct the ICS test at medium power. Testing at high power would not be more challenging from the viewpoint of the structural integrity of the ICS. Hence, no DCD change is required.

The staff finds the response to RAI 14.2-3 acceptable.

The ICS tested at the PANTHERS-IC facility was one module of a full-scale, two-module ICS in the ESBWR. The staff concludes that the test results using one module demonstrated the capability of a prototype ICS module to perform as designed with respect to heat rejection and provided a database for TRACG qualification regarding the quasi-steady heat removal rate of an ICS. The PANTHERS/ICS data are, therefore, acceptable as a valid database to qualify the TRACG code for the ICS global heat removal rate.

Depressurization Valve (DPV) Tests

Researchers conducted full-size DPV tests at the Wyle Laboratory in the United States. The test objective was to demonstrate reliable operation of the DPV.

Mass flow rate in a DPV was not measured because the tests focused on the successful opening of the DPV. GEH conducted full-size testing of the DPV to demonstrate its operation and reliability.

In RAI 3.9-1, the staff requested GEH to submit the test reports for the DPVs. Section 3.9 of this report presents the staff's evaluation of the DPV test results.

Vacuum Breaker Tests

Researchers conducted full-size VB tests at a GEH facility in the United States. The test objective was to demonstrate reliable operation of the VB.

The opening and closing pressures of a VB were measured. GEH has conducted full-size VB tests to demonstrate its operation and reliability.

Also, in RAI 3.9-1, the staff requested GEH to submit the VB test reports. Section 3.9 of this report provides the staff's evaluation of the VB test results.

21.5.3.2 <u>Integral Systems Tests</u>

Integral systems tests were conducted at the GIST, GIRAFFE, and PANDA test facilities.

Gravity-Driven Integrated Systems Tests

The test objectives were to demonstrate the technical feasibility of the GDCS concept and to provide a sufficient database to confirm the adequacy of the TRACG code in predicting GDCS flow initiation times, GDCS flow rates, and RPV water levels.

The gravity-driven integrated systems tests (GIST) focused on the ability of the GDCS to maintain core cooling in a LOCA and were performed by GEH in San Jose, California, in 1988. The GIST facility was a section-scaled simulation of the 1987 SBWR design configuration, with a 1:1 vertical scale and a 1:508 horizontal area scale of the RPV and containment volumes. Because of the 1:1 vertical scaling, the tests provided real-time response of the 1987 SBWR pressures and temperatures.

The GIST test program included the effects of various plant conditions on GDCS initiation and performance. The GIST facility consisted of four pressure vessels—the RPV, upper drywell, lower drywell, and wetwell. The wetwell included the GDCS fluid. The RPV included internal structures, an electrically heated core, and bypass and chimney regions.

The GIST facility modeled the SBWR plant behavior during the late stage of the RPV blowdown. The tests were started with the RPV at 791 kilopascals (kPa) (100 pound-force per square inch gauge (psig)) and continued until the GDCS flow initiated and flooded the RPV. Four types of tests were conducted—MSLB, GDLB, BDLB, and a no-break scenario (e.g., loss of feedwater). All these tests lasted from 600 to 1210 seconds. Researchers conducted 29 integral systems tests.

Investigators evaluated the integral systems response of the RPV and containment during the late blowdown phase and GDCS injection phase of LOCAs.

Unlike the PANDA M-series and GIRAFFE tests, the GIST tests were conducted in a facility that was based on an older SBWR design that did not include a separate GDCS pool. Instead, the elevated suppression pool (SP) also served as the GDCS coolant source. In this respect, the GIST design was closer to that of the ESBWR.

Three kinds of LOCAs were tested in GIST—an MSLB, GDLB, and BDLB. Sensitivity studies performed by GEH at that time indicated that these breaks were expected to bracket other LOCAs in terms of break sizes, locations, and coolant flow. Nineteen LOCA tests were conducted, which included eight MSLB tests, four GDLB tests, and seven BDLB tests. For the same kind of LOCA (e.g., the MSLB), initial test conditions were varied among the reactor vessel water level, SP level, and the number of operational GDCS injection lines. The figure of merit, the critical safety parameter, for the GIST tests was the minimum downcomer water level.

The tests demonstrated the technical feasibility of depressurizing the RPV to sufficiently low pressures below the static head of an elevated pool of water in the containment, enabling coolant injection to the core.

Design limitations caused two phenomenon distortions. First, GIST used two vertical pipes as the replacement for the annular downcomer of the reactor vessel between the lower plenum and the upper plenum above the core. Asymmetrical behavior observed during part of the tests revealed a two-phase or frothy mixture in one downcomer pipe and phase separation (low-void water in the bottom with steam above) in another downcomer pipe. This kind of asymmetry is not expected to occur in the annular vessel downcomer of the ESBWR since it does not have the separation found in the test facility's separate downcomer pipes. Second, a single standpipe was installed above the upper plenum of the RPV, where periodic percolation was identified during part of the tests, which led to periodic variations in the RPV pressure. However, these distortions are nonprototypical and are not expected to invalidate the overall integral systems behavior observed in the GIST tests.

The staff concludes that the GIST tests demonstrated the technical feasibility of the GDCS concept, which involves RPV depressurization to allow coolant injection to the vessel from an elevated pool of water in the containment. Despite the phenomenological distortions described above, the GIST tests demonstrate that the overall GDCS performance in providing coolant to a depressurized RPV remains valid for a broad spectrum of LOCAs. The GIST data are therefore acceptable as a valid database to qualify the TRACG code for the late blowdown and early GDCS injection phases of a LOCA in the ESBWR.

GIRAFFE/Helium Tests

The test objectives were to (1) demonstrate the operation of a PCCS with the presence of a lighter-than-steam noncondensible gas, including the process of purging noncondensible gases from the PCCS, (2) provide a database to confirm the adequacy of TRACG to predict SBWR containment system performance in the presence of a lighter-than-steam noncondensible gas, including potential systems interaction effects, and (3) provide a tie-back test, which includes the appropriate QA documentation, to repeat a previous GIRAFFE test.

GIRAFFE/Helium tests were performed as a joint effort by GEH and Toshiba in Kawasaki City, Japan. The GIRAFFE facility is a large-scale, integral system test facility designed to exhibit post-LOCA thermal-hydraulic behavior similar to the SBWR systems that are important to long-term containment cooling following a LOCA.

The global volume scaling of the facility is approximately 1:400, with a nominal height scaling of 1:1. The SBWR components simulated in the facility are the RPV, PCCS, GDCS, drywell, wetwell, and connecting piping and valves. Five separate vessels represent the SBWR RPV, drywell, wetwell, GDCS pools, and PCCS pool. The facility was equipped with one PCCS to represent the three SBWR PCCS condensers. Electric heaters provided a variable power source to simulate the core decay heat and the stored energy in the reactor structures.

For the helium series tests, once the initial test conditions were established, all control (except for the decay of RPV power and helium injection, if called for) was terminated, and the GIRAFFE containment was allowed to function without operator intervention (except for the VB, which was operated manually to simulate automatic operation in the SBWR, and the minor wetwell microheater power adjustments that were made to compensate for facility heat losses).

In the GIRAFEE/Helium tests, the phenomenon investigated was the integral system response of the RPV and containment during the long-term cooling phase of LOCAs. Researchers conducted four tests to demonstrate the PCCS operation with the presence of a lighter-than-steam noncondensible gas (using helium as a substitute for hydrogen gas) and a heavier-than-steam noncondensible gas (nitrogen). Test H1 was the base case test, and the initial test conditions were based on TRACG calculations for the SBWR during the long-term cooling phase at 1 hour after the break initiation (RPV initial pressure at 295 kPa or 42.8 psia). Test H2 was a repeat of test H1, but with helium replacing the nitrogen in the drywell. Test H3 was a variation of test H1, but with helium replacing some steam in the drywell. Test H4 was similar to test H1, but with a constant helium injection into the drywell. In addition, two other MSLB tests, tests T1 and T2, were conducted with nitrogen as the only noncondensible gas in the containment.

Heat loss was a concern in the GIRAFFE facility, which was tall and thin. Electric microheaters were installed to wrap around the metal walls of the drywell, wetwell, and GDCS pool, which were covered with an insulation material. Microheater power for each component was determined during the shakedown tests to compensate for the heat loss. Since the microheater power could not fully compensate for the heat loss, the RPV electric heater power was raised above the scaled decay heat to further compensate for the heat loss in the facility with the microheaters on. But this provision could not eliminate the local heat loss in the lower drywell, which was found to be significant. The heat loss has the potential to introduce some local distortions in the test data, and therefore, should be taken into consideration for code uncertainty evaluation.

There were only two noncondensible gas sampling locations in the drywell—one at the top of the drywell and the other at the very bottom of the drywell located in the lower drywell where the local heat loss was significant. The heat loss at the bottom sampling location has the potential to somewhat distort the noncondensible gas behavior in the drywell. This problem was compounded by the scarcity of the noncondensible sampling locations. For the wetwell gas space, there was only one noncondensible gas sampling location. However, unlike the lower drywell, the wetwell wall heat loss was found to be insignificant. The scarcity of the noncondensible gas sampling locations and the heat loss problem at the lower drywell tended to reduce the quality of the containment noncondensible gas distribution data. These limitations of test data were overcome by employing conservatively bounding TRACG containment models.

All the GIRAFFE/Helium tests (including tests T1 and T2) focused on the long-term cooling phase of the MSLB and did not include the late blowdown and GDCS phase. The GIRAFFE/Helium tests demonstrated the ability of the PCCS to maintain containment cooling during the long-term cooling phase of the MSLB, which was the most critical LOCA to challenge the containment for the SBWR. Investigators evaluated the impact on the PCCS performance for both heavier-than-steam (nitrogen gas) and lighter-than-steam (helium gas) noncondensible gases present in the containment under various test conditions.

Because of the heat loss at the lower drywell, noncondensible gas distribution in the drywell is distorted by having a much higher noncondensible concentration (because of local steam condensation) than expected in the lower drywell. Furthermore, since there were only two noncondensible sampling locations in the drywell and only one in the wetwell gas space, extra efforts were needed to interpret and use the data to qualify the TRACG code with regard to the noncondensible gas distributions in the containment. Nevertheless, the many measurements of pressures, temperatures, and water levels were sufficient to explain the containment response in the presence of the heavier-than-steam and lighter-than-steam noncondensible gases.

The GIRAFFE/Helium tests were based on the SBWR design, which is very similar to the ESBWR design in terms of the RPV and containment phenomena expected in a LOCA. Furthermore, the design changes from the SBWR to the ESBWR did not introduce any new phenomena. In view of the above, the staff concludes that the GIRAFFE/Helium tests provided a valid database to qualify the TRACG code for the long-term cooling phase of a LOCA involving both lighter-than-steam and heavier-than-steam noncondensible gases, although a careful examination of all the data was necessary.

GIRAFFE Systems Interactions Tests

The test objective was to provide a database to confirm the adequacy of TRACG to predict the SBWR ECCS performance during the late blowdown phase and GDCS injection phase of a LOCA, with specific focus on potential systems interaction effects.

Researchers conducted a series of four transient systems tests to provide an integral systems database for potential systems interaction effects in the late blowdown and GDCS injection phases. All four tests involved liquid breaks—three GDLBs and one BDLB. Tests were performed with and without the ICS and PCCS in operation and with two different single failures.

The tests investigated the post-LOCA thermal-hydraulic behavior (especially the RPV pressure transient and water level transient), the GDCS injection characteristics, and possible systems interactions. The test facility modeled the whole containment system of the SBWR. The SBWR components modeled in the facility were the RPV, ICS, GDCS, PCCS, drywell, wetwell, and connecting piping and valves. Major portions of the SBWR containment (drywell, wetwell, and GDCS pool, as well as the ICS and PCCS pools) were modeled using separate vessels.

The PCCS unit was the same as that used for the GIRAFFE/Helium tests and consisted of a steam box, heat transfer tubes, and a water box. The PCCS had three heat transfer tubes corresponding to the scaled volume. The heat transfer tubes were full height, and the internal tube flow area was almost the same as the scaled SBWR flow area. One scaled ICS was mounted above the drywell vessel. The ICS had three tubes, two of which were plugged to reduce the heat transfer surface of the unit. This single condenser represented the two ICS condensers found in the SBWR.

Testing followed a methodology very similar to that used in the PANDA and GIRAFFE/Helium tests. Once the initial conditions for a given test were established, all controls (except for the decay of RPV power) were terminated. The GIRAFFE RPV and containment were allowed to function without operator intervention. The GDCS pool-to-drywell flow was manually terminated at 1 hour in the GDCS break cases to avoid an inappropriate emptying of the pool. This was necessary since a single pool in the GIRAFFE simulated the three SBWR pools, only one of which would have pool-to-drywell flow. Manually stopping GDCS flow to the drywell in the GIRAFFE tests simulated the end of draining for that one pool in the SBWR and maintained the simulation of flow from the remaining pools to the RPV.

Phenomena associated with the integral systems tests were investigated. Integral systems responses of the RPV and containment in the late blowdown and GDCS injection phases of the GDLB and BDLB were measured. By comparing two similar GDLB tests with and without PCCS and ICS operation, investigators could assess interactions between the PCCS/ICS and GDCS.

Four integral systems tests were conducted to assess the GDCS performance in maintaining a covered core with and without the operation of the ICS and PCCS. Two kinds of LOCAs were investigated with break locations below the main steamline elevation—GDLB and BDLB. Test GS1 comprised a GDLB without the operation of the PCCS and ICS, assuming a DPV failure (failed to open upon demand). Test GS2 was similar to test GS1, but included the operation of the PCCS and ICS. Test GS3 was a BDLB with the operation of the PCCS and ICS, assuming a DPV failure. Test GS4 was a GDLB with the operation of the PCCS and ICS, assuming a valve failure on a GDCS injection line. These tests complemented the GIRAFFE/Helium tests in which only the MSLB was investigated. Potential interactions between the GDCS operation and the PCCS/ICS operation were assessed.

The GIRAFFE heat loss problem, discussed in the GIRAFFE/Helium tests, was also present in the GIRAFFE systems interactions tests. Although electric microheaters were used around the drywell, wetwell, and GDCS pool, and the RPV heater power was increased beyond the scaled decay heat to compensate for the heat loss, the heat loss problem could not be fully eliminated. For instance, the local heat loss in the lower drywell was found to be significant. As indicated earlier, heat loss has the potential to introduce some local distortions in the test data, and therefore, should be taken into consideration for code uncertainty evaluation.

GIRAFFE/systems interactions tests lasted only 2 hours, which was not long enough to lead to the potential opening of the equalizing lines to provide SP water to the RPV. As a result, the equalizing line mass flow was not observed in the test data.

In all four tests conducted, the GDCS injection ran smoothly without noticeable flow oscillations. It performed well in keeping the core covered and maintaining core cooling. Comparing tests GS1 and GS2, the PCCS/ICS operation had no adverse impact on GDCS performance and led to a lower containment pressure as expected. Operation of the ICS significantly reduced the steamflow available to the PCCS, except for the initial 200 to 300 seconds.

In RAI 21.5-1, the staff requested GEH to clarify, in the DCD, the importance of the SP equalization line for long-term cooling, particularly the long-term PIRT ranking of the equalization line, and if necessary, to describe appropriate testing.

In response, GEH stated that the equalization line valves are not expected to open for a LOCA resulting from a break in any of the lines in the current ESBWR design, as submitted in the DCD. The results for the downcomer level response for the first 12 hours following a BDLB, FWLB, GDCS line break, and MSLB showed that the downcomer water level stabilized at an elevation well above the elevation of the L0.5 trip (1 meter above the top of active fuel and approximately 8.5 meters above the bottom of the RPV). The lowest level in the long term occurs for the GDCS line break, which still has more than a 1-meter margin to L0.5. This resulted from two changes in the current design relative to the previous design analyzed— (1) a larger GDCS pool volume and (2) a smaller lower drywell volume.

GEH further stated that Table 2, “Containment/LOCA long term PIRT,” in MFN 05-109, “GE Response to Results of NRC Acceptance Review for ESBWR Design Certification Application - Item 2,” October 20, 2005, showed an incorrect “High” ranking for equalization line friction (EQ1). This table was extracted from a previous report, which did not reflect the changes in the ESBWR design mentioned above. Therefore, the ranking for SP equalization line (EQ1) in the PIRT should be “N/A” (i.e., not applicable), because the equalization line valves are not expected to be activated for any design-basis events in the current ESBWR design. The staff finds the GEH response acceptable, and therefore, no additional testing is required.

The GIRAFFE/Helium tests were based on the SBWR design, which is very similar to the ESBWR design in terms of the RPV and containment phenomena expected in a LOCA. Furthermore, no new phenomena were introduced as a result of the design changes from the SBWR to the ESBWR. Accordingly, the staff concludes that the GIRAFFE systems interactions tests provide a valid database to qualify the TRACG code for the late blowdown and GDCS injection phases of a LOCA.

PANDA M-Series Tests

The test objectives were to (1) provide a sufficient database to confirm the capability of TRACG to predict SBWR containment system performance, including potential systems interaction effects, and (2) demonstrate startup and long-term operation of a PCCS.

PANDA M-series tests were performed as a joint effort by GEH and PSI in Wuerenlingen, Switzerland. The test facility was a large-scale integrated containment structure which was a 1/25-volumetric, full-height, scaled model of the SBWR containment. It was a modular facility with separate pressure vessels representing the RPV, drywell, wetwell, and GDCS pool. The facility was equipped with three scaled PCCS heat exchangers and one ICS unit (scaled from two SBWR ICS units), each with a separate pool of water. Electrical heaters were used in the RPV to simulate decay heat and the thermal capacitance of the RPV walls and internals in the SBWR. The test facility also had interconnecting piping arrangements needed to conduct the MSLB tests. The tests were started at an equivalent condition from about 1040 seconds (transition from the GDCS injection phase to the long-term cooling phase) to about 3600 seconds (beginning of the long-term cooling phase) after the initiation of the MSLB in the SBWR. The duration of a test was up to 20 hours.

When the initial conditions for a given test were established, all controls were terminated except for automatic control of the wetwell-to-drywell VB position and the electric heater simulation of the RPV structure stored energy release and core decay heat power. The PANDA containment was then allowed to function without operator intervention. The only exceptions to the procedure described above were for tests M3A and M3B, which included operator action to maintain PCCS pool level, and test M6/8 during which the operator established a drywell-to-wetwell flowpath (bypass leakage) and later valved the ICS unit out of service.

Integral systems response of the RPV, drywell, and wetwell was investigated for the late GDCS injection phase and long-term cooling phase of an MSLB LOCA. PCCS performance for maintaining containment cooling was assessed.

PANDA was a “large” test facility at a scale of 1/25 of the SBWR. It had all the necessary components to conduct the integral systems tests to investigate the long-term cooling phase of a DBA, namely the MSLB accident which was expected to be the most challenging LOCA to the containment for the SBWR.

The PANDA M-series tests consisted of 10 integral systems tests for the MSLB that covered a broad spectrum of test conditions expected in the SBWR. Except for test M9, these tests focused on the long-term cooling phase of the MSLB (occurring at about 1 hour after break initiation). Test M9 included both the late GDCS injection phase (with the initial test conditions based on 1040 seconds after the break initiation in the SBWR) and the long-term cooling phase of a LOCA. These tests demonstrated successful operation of the PCCS for maintaining adequate containment cooling under various MSLB conditions in a large test facility.

PANDA M-series tests were designed to focus on the MSLB accident because that was expected to be the most challenging LOCA to the containment for the SBWR. There was no lower drywell in the PANDA test facility, and consequently, the GDLB and BDLB could not be tested. Potential opening of the GDCS equalizing lines to provide SP water to the RPV could not be investigated. (See previous discussion regarding RAI 21.5-1.)

The volume of the GDCS pool was much smaller than the scaled volume, and consequently there was an insufficient amount of water to cover the entire spectrum of the GDCS injection phase. As a result, the PANDA tests investigated the long-term cooling phase and only a portion of the GDCS injection phase of the MSLB LOCA. Because the primary objectives of the test were to investigate long-term containment phenomena and not the GDCS injection phase, the staff finds this acceptable.

Large oscillations occurred in the main steamline mass flow rates, when the water level in the RPV was high (close to the top of the chimney). The flow oscillations were greatly reduced if the initial RPV water level was at a low level (several meters below the top of the chimney). The staff believes that the flow oscillations might have been caused by design distortions in the PANDA test facility (e.g., lack of core inlet orifices, fuel assemblies, steam separators, dryers, and multiple fuel assemblies in the RPV) although they did not prevent the PCCS from maintaining containment cooling.

The PANDA test facility had all the necessary components to conduct the integral systems tests for a design-basis LOCA such as the MSLB. The M-series tests covered a broad spectrum of the test parameters expected in the SBWR (which are similar to the ESBWR test parameters) to investigate the long-term cooling phase of a LOCA. The PCCS performed well, maintaining adequate containment cooling in the MSLB test. Drywell air was purged to the wetwell by means of the PCCS. There was a smooth transition from the GDCS injection phase to the long-term cooling phase. The VB openings in a test did not significantly affect the global drywell and pressure response, as compared to a similar test without the VB openings.

Although the PANDA M-series data are for the MSLB test conditions, the containment phenomena in the long-term cooling phase of other LOCAs, such as the GDLB, BDLB and FWLB, are generally similar to those of the MSLB (with an exception to be discussed below). This is because, before the start of the long-term cooling phase (with variations in the starting time that is LOCA dependent), the RPV has depressurized from the ADS actuation and the GDCS injection has become insignificant. However, there was one exception. As stated above, the potential opening of the GDCS equalizing lines to provide SP water to the RPV could not be investigated in the PANDA test facility. (See previous discussion regarding RAI 21.5-1.)

As stated earlier, the PANDA M-series tests were based on the SBWR design, which is very similar to the ESBWR design in terms of the RPV and containment phenomena expected in a LOCA. Furthermore, the design changes from the SBWR to the ESBWR did not introduce new phenomena. Equally important, the phenomena observed in the PANDA M-series tests were generally understood and appeared to be reasonable. For example, the addition of relatively cold water at room temperature to the PCCS pools temporarily enhanced the overall PCCS heat removal rate and could lead to VB opening. But this did not significantly affect the overall behavior of the drywell and wetwell pressures. Therefore, the staff concluded that the PANDA M-series tests provided a valid database to qualify the TRACG code for the long-term cooling phase of a LOCA relevant to the ESBWR LOCA events.

PANDA P-Series Tests

The test objectives were to (1) reinforce the existing database to confirm the adequacy of TRACG to predict the ESBWR containment performance, including potential systems interaction effects, and (2) confirm the performance of an earlier preapplication version of the ESBWR containment configuration with the GDCS gas space connected to the wetwell gas space.

In the current ESBWR design as submitted for design certification, GEH modified the design by moving the connections of the GDCS pool airspace from the wetwell back to the drywell and eliminating the connecting vent between the wetwell airspace and the GDCS pool airspace. Therefore, this configuration is the same as the arrangement in the SBWR design and in the integral systems test programs, PANDA M-series and GIRAFFE, used for qualification of the TRACG code. Containment volumes were adjusted along with this change to ensure the wetwell-to-drywell volume ratio and thus retained most of the benefit of the reduced containment pressure that was gained when this GDCS airspace volume was originally moved from the drywell to the wetwell. While the earlier (preapplication version) ESBWR configuration provided additional margin in the containment pressure performance, it resulted in several complicating design issues necessitating GEH to implement this modified ESBWR configuration, which is similar to the original SBWR configuration.

PANDA is a large-scale integral test facility originally designed to model the long-term cooling phase of a LOCA for the SBWR. It has all the major components, including the RPV, drywell, wetwell, and GDCS pool. The RPV was equipped with electrical heaters and heater controls to simulate decay heat and the release of RPV stored energy. The facility included all three scaled PCCS heat exchangers and one ICS unit and their associated water pools. Other components represented in PANDA include VBs between the drywell and the wetwell and the equalizing lines between the SP and the RPV.

The RPV was modeled using a single vessel in PANDA, while the drywell and wetwell were modeled using two pairs of vessels, connected by large pipes. This double-vessel arrangement permitted investigation of spatial distribution effects within the containment volumes. The water in the RPV was heated by a bank of controlled electrical heaters that could be programmed to match the decay heat curve. Main steamlines conveyed boil-off steam from the RPV to the two drywell vessels. The PCCS and ICS inlet lines were connected to the drywell and RPV, respectively. Drainlines from the lower headers of the PCCS and ICS units returned condensate to the RPV. Ventlines from the lower headers of the PCCS and the upper and lower headers of the ICS were at prototypical submergences in the SP. VBs were located in the lines connecting the drywell and wetwell gas spaces. PANDA had the capability to valve out one of the main steamlines, the ICS, and individual PCCS. It also had the capability to inject noncondensible gas (air or helium) into the drywell over a prescribed time period during the post-LOCA transient tests.

As stated above, in the original PANDA/SBWR configuration (for the PANDA M-series tests), the GDCS gas space was connected to the drywell. A major modification made in the PANDA/SBWR was to connect the GDCS gas space to the wetwell gas space (for the PANDA P-series tests) to model a preapplication version of the ESBWR configuration. This ESBWR configuration, which was not adopted as the final ESBWR design, provided a larger volume for the noncondensible gases that are purged from the drywell to the wetwell during the blowdown phase and therefore reduced the containment pressure. In its original configuration for the SBWR, PANDA was a 1/25-volume-scaled, full-height representation of the SBWR primary system and containment. As configured for the P-series tests for the ESBWR, the PANDA facility was a full-height representation of the ESBWR containment at a nominal volumetric scale of 1:45. The piping interconnecting the PANDA vessels was scaled (primarily with the use of orifice plates) to produce the same pressure loss as the corresponding ESBWR piping. The three PANDA PCCS units were approximately equivalent to the four ESBWR PCCS units, and the one PANDA ICS unit was about 10 percent underscaled relative to the four ESBWR ICS units.

The tests investigated integral systems response of the RPV, drywell, and wetwell for the late GDCS injection phase and the long-term cooling phase of the MSLB. PCCS performance for maintaining containment cooling was also assessed.

As stated earlier, the PANDA P-series tests were based on a preapplication version of ESBWR configuration, in which the GDCS pool was isolated from the drywell and its gas space was connected to the wetwell gas space instead of the drywell, as in the SBWR and the current ESBWR configuration. In addition, the PCCS drainlines were connected to the RPV instead of the GDCS pool, as in the SBWR and the ESBWR. The P-series tests consisted of eight integral systems tests for the MSLB (which was expected to be the most challenging LOCA to the containment for the SBWR) to investigate the containment response and phenomena during the long-term cooling phase under various initial and boundary conditions. PCCS performance was successfully demonstrated to maintain containment cooling. Various containment phenomena were investigated. The changes noted made the PANDA-P tests consistent with the preapplication version of the ESBWR configuration with minor deviations.

Like the PANDA M-series tests, the PANDA P-series tests were conducted in the same facility except with modifications necessary to conform to a preapplication ESBWR configuration as stated earlier. There was no lower drywell, and other LOCAs with a lower break location, such as the GDLB and BDLB, could not be tested. Tests did not investigate potential openings of the SP equalizing lines to provide SP water to the RPV.

The PCCS pools in PANDA were much smaller than the scaled volume. For tests longer than about 35,000 seconds (9.7 hours), the PCCS condenser tubes were uncovered unless water was added to the pool from an outside source.

The PANDA facility has all the necessary components to conduct the integral systems tests for a design-basis LOCA, such as an MSLB. The P-series tests covered a broad spectrum of the test conditions expected in the ESBWR to investigate the long-term cooling phase of a LOCA. The PCCS performed well and maintained adequate containment cooling in the MSLB tested. The transition was smooth from the late GDCS injection phase to the long-term cooling phase. Injection of a noncondensible gas (using either air to simulate nitrogen or helium to simulate hydrogen) to the drywell degraded the PCCS performance. The PCCS was capable of purging noncondensible gas from the drywell to the wetwell, as it was injected.

At a low decay heat equivalent to several hours into the MSLB, the test data suggested that the PCCS was capable of maintaining containment cooling even when the PCCS condenser tubes were substantially uncovered.

Although the PANDA P-series data are for the MSLB application, the containment phenomena in the long-term cooling phase of other LOCAs, such as the GDLB, BDLB, and FWLB, are generally similar to those of the MSLB. The reason is, before the start of the long-term cooling phase, the RPV has depressurized from the ADS actuation. As stated earlier, the PANDA tests could not investigate the potential opening of the SP equalizing lines to provide SP water to the RPV. (See previous discussion regarding RAI 21.5-1.)

Some of the data have revealed distortions (e.g., a temperature rise in the wetwell gas space from nonprototypical heating from the gas flow in the vertical main vent pipe until it was valved out). These nonprototypical distortions are not expected to change the overall containment behavior. The phenomena observed in the PANDA P-series tests are generally understood and seem to be reasonable. For example, when a VB opened, some of the wetwell noncondensible gas flowed to the drywell and degraded the PCCS performance. As a result, the drywell pressure first rose and eventually leveled off when the pressure difference between the drywell and the wetwell was sufficient to overcome the PCCS vent submergence and vent pipe flow resistance. As expected, main vents cleared (to vent the drywell gas directly into the wetwell) when there was insufficient heat removal in the PCCS as a result of either the absence of one PCCS unit (out of a total of three) or noncondensible gas injection to the drywell during a test.

On the basis of the discussion made above, the staff concluded that although having a difference in containment configuration between the PANDA P-series tests and the current ESBWR, the PANDA P-series tests provided a valid database to confirm the qualification of the TRACG code for the long-term cooling phase of a LOCA relevant to the ESBWR LOCA events; in particular, it provided data on PCCS performance with noncondensible gas at an additional scale.

21.5.3.3 Summary of the ESBWR Component and Integral Systems Testing Programs

The results of the single tube condensation test program performed at UCB were the basis for the condensation heat transfer correlation used in the TRACG code. The full-size component test data from the PANTHERS/PCCS and PANTHERS/ICS test programs cover the range of the operational conditions expected in the design-basis LOCAs in the ESBWR. These data are adequate for validating the TRACG code regarding the PCCS and ICS performance in the ESBWR (with the understanding that a PCCS condenser in the ESBWR has approximately 35 percent more heat removal capability than does the PANTHERS/PCCS condenser and an ICS condenser has twice the heat removal capability as the single-module PANTHERS/ICS condenser).

The integral systems test data from the GIST, GIRAFFE/Helium, GIRAFFE Systems Interactions, PANDA M-series, and PANDA P-series testing programs as a whole cover a range of the late blowdown phase, GDCS phase, and long-term cooling phase of the accidents. The staff understood the phenomena revealed in the data and concluded that the weaknesses (including some phenomenon distortions) in general do not invalidate the overall reactor vessel and containment response in a LOCA. The combined data from the GIST, GIRAFFE, and PANDA integral systems tests covered the LOCA phenomena and processes defined in the PIRTs for the late blowdown phase, GDCS phase, and long-term cooling phase.

Each integral systems test provided a set of “valuable” data on the time-dependent, thermal-hydraulic response of the RPV, drywell, and wetwell with the operation of the GDCS, PCCS, or ICS in a LOCA. For the TRACG code to properly simulate the test, the code must have technically sound conservation equations, including the constitutive package and numerics. As a result, the data of an integral systems test are useful for assessing a code against the test for the specific test configuration and initial and boundary conditions. However, to link the integral systems test data to the ESBWR response in a LOCA required an adequate scaling analysis to demonstrate the applicability of the test data to the ESBWR response. GEH performed such a scaling analysis, and the staff evaluated it, as discussed below.

In conclusion, the staff has reviewed and evaluated the test programs performed originally in support of the GEH SBWR design and found the testing to be applicable to the ESBWR design, based on the PIRT and scaling analysis as discussed below. Based on the design description for the ESBWR provided in the DCD, the staff also concludes that no further testing in support of LOCA thermal-hydraulic behavior of the design is necessary.

21.5.3.4 Determination of Effect of Scale

Various physical processes may give different results as components or facilities vary in scale from small to full size. The quantification of bias and deviation must include the effect of scale to determine the potential for scale-up effects.

GEH used the hierarchical two-tier scaling (H2TS) process. One of the key elements of the H2TS approach is the identification of the important physical phenomena governing a process. Generally, the phenomena are identified and ranked in importance, and the results of this effort are documented in a PIRT table. The H2TS approach consists of a top-down method, which is a system scaling analysis used to derive scaling groups and establish a scaling hierarchy, and a bottom-up method, which focuses on the important processes and introduces similitude to ensure that the scaled test data are applicable to the prototype. The top-down system scaling does not replace, but rather provides a rational framework for the bottom-up scaling. NUREG/CR-5809, *A Hierarchical Two-Tiered Scaling Analysis*, November 1991 describes the H2TS approach.

Evaluation of the GEH Scaling Analyses

To evaluate the adequacy of the GEH scaling approach, the objectives of a scaling analysis for code assessment were defined, and that definition was used to evaluate how the GEH ESBWR scaling report NEDC-33082P, "ESBWR Scaling Report," Class III, December 2002, (NEDC-33082P) demonstrated that the objectives were accomplished. NEDC-33082P defined the objective as "to show that the test facilities properly 'scale' the important phenomena and processes identified in the ESBWR PIRT and/or provide assurance that the experimental observations from the test programs were sufficiently representative of ESBWR behavior for use in qualifying TRACG for ESBWR design basis calculations." The staff accepted the objective as stated in the GEH report.

GEH adopted the H2TS approach for the ESBWR. The LOCA served as the basic event for the scaling analysis. Since the importance of the governing phenomena changes as the event unfolds, GEH defined four accident phases that span the accident, namely, late blowdown, GDCS initiation, GDCS phase, and PCCS phase. The early blowdown period is not significant for passive safety system performance and was therefore ignored. The primary test facilities scaled for SBWR and ESBWR testing can simulate decay power levels starting at approximately 1 hour after the initiation of the accident. Since a key issue is PCCS performance, the scaling was directed at the late blowdown phase extending into the long-term cooling phase. The long-term cooling phase is unique to the SBWR/ESBWR containment because of the substitution of passive for active cooling systems.

GEH began its scaling efforts with a PIRT. The top-down scaling approach complements the PIRT by identifying the important phenomena during each accident phase based on nondimensionalization of the governing equations. The global momentum and energy conservation equations used were based on the lumped-parameter approach. The system was divided into several large volumes. The equations of energy and mass balance developed for a generic volume were then applied to each of these volumes at different time periods during the transient. The equations were made nondimensional and the resulting nondimensional coefficients were defined as the Pi's to represent the relative importance of the participating phenomena.

The bottom-up scaling considered the individual phenomena at a local level. GEH used bottom-up scaling to look at specific processes important to system behavior in more detail. For ESBWR, 46 highly ranked phenomena needing detailed evaluation were identified and provided the basis for acceptability of the data for TRACG qualification.

The main objective of integral scaled facilities was to capture not only the component behavior, but also its dynamic interactions as a complete system. NEDC-33082P acknowledged this in the executive summary, which stated that, "A comprehensive experimental program was carried out to demonstrate the thermal-hydraulic performance of these passive systems and their components." The staff, however, noted that the analysis presented in the report did not account for systems interactions. The staff believes that, while one cannot expect that any of the scaled facilities represent a simulation of the prototype, for completeness, they must at least exhibit the same kind of interactions between components and subsystems as expected of the prototype. It is up to the scaling analysis, therefore, to determine the relevancy of these interactions. System interactions are not explicitly called out in the PIRT as phenomena. They are, however, an integral part of the transient, and they determine the sequence of events that define the beginning of a phase, the end of a phase, and what process controls the state of the system during that phase.

In general, the reactor system was divided into subsystems for which governing equations were developed. The governing equations were made nondimensional by referring all variables to a set of norms or reference parameters (including a reference time), according to the purpose of the analysis. The intent of this process was to obtain nondimensional parameters. The nondimensional coefficients of these equations, the system Pi's, contain information about how the different components of the system interact and which of these many interactions dominates the transient behavior during a given phase.

During each transient, the system state and its configuration changes as the transient progresses from one phenomenologically distinct phase to the next. Each of these phases will include a process or a set of competing processes that define the beginning and the end of the phase and therefore its reference time. The general approach needs to be repeated for each system configuration and each reference time.

The top-down scaling should reach a certain level of system detail. At one extreme, the approach could assume that the entire reactor system is one comprehensive volume and conduct the analysis accordingly. The result would be simple and of limited value. At the other extreme, the approach would call for as much detail as possible, without invoking multidimensional effects or the local distribution of a phenomenon. The latter would likely result in a system representation that varies from phase to phase of the transient, as the system configuration varies (valves open and close, tanks empty or fill).

GEH selected an in-between approach and identified the major system volumes as the components, all represented, in principle, by the same equations of energy and mass conservation. The momentum equations of the connecting lines or paths were neglected as having no dynamic contribution. Furthermore, NEDC-33082P, Section 6.2, cited previous efforts by stating, “results from the SBWR work showed that there are no significant interactions in the SBWR system or the related tests and no new Pi numbers resulted.” The staff, however, believes that the SBWR study presented in NEDC-32288P found that the lines and connecting paths have very fast response times compared to other simultaneous processes and that they contribute enough damping to suppress oscillations. The last paragraph of Section 6.2 suggested that the analysis conducted for the SBWR was not carried out for the ESBWR because the designs are “similar enough.” However, the staff believes that in both the SBWR and ESBWR, the volumes do interact because they are connected. In response to the staff request, GEH addressed this deficiency by performing a revised scaling analysis as discussed later in this section.

The statement in NEDC-33082P, Section 6.3, “these equations are applied to the specific regions of the ESBWR,” raised the question in the staff’s mind that the GEH original scaling approach ignored interactions. Even when there were two or three volumes actively participating and interacting with each other, the GEH approach addressed the volumes independently in NEDC-32288P and NEDC-33082P. The staff believes that the volume equations (mass and energy) have terms that represent inflows and outflows. In most cases, these are not external inputs to the reactor system, but result from gradients between connecting volumes and, therefore, are not independent variables. A single volume equation can neither capture nor describe this system behavior and is insufficient to draw conclusions about that behavior. It is likely that the two or three volumes involved were interdependent and could be represented by a single equation. As a result, the staff concluded that the equation used by GEH in its analysis was not capable of demonstrating system interactions. In fact, the GEH original scaling approach in NEDC-32288P and NEDC-33082P considered no analysis of system interactions at all.

The nondimensional coefficients, or Pi groups, identified in the top-down scaling are more complex than the more traditional similarity parameters derived in the study of physical phenomena, such as the Reynolds number and Prandtl number. Evidence of this complexity is the fact that a characteristic system time is an integral part of these Pi groups and they come in sets of two or more. The Pi groups are derived from the macroscopic analysis of distinct elements of the system that accounts for the way in which the elements interact and exchange mass, energy, or both with each other and with the environment. These Pi groups are a useful tool to determine what processes or mechanisms dominate the behavior for each particular system. They can also be used to assess whether two different systems can be expected to have similar behavior. However, the similarity can only be guaranteed a priori if the two systems have identical Pi groups. If the Pi group values differ, further analysis is necessary to assess the similarity between the different systems. The most important part of this further analysis is the verification that the data—and code calculation for the test facility—exhibit the same trends, magnitudes, and variations in nondimensional space. The other aspect of this analysis is the evaluation of local phenomena to ensure that, while the systems are expected to be similar in their macroscopic behavior, the local phenomena (bottom-up) support this expectation by producing the same regime. This invokes the more traditional nondimensional groups, such as Reynolds, Prandtl, and Biot numbers, which correspond to the local processes not captured by the top-down formulation of the system equations. The GEH scaling report, NEDC-33082P, in its original version, was very weak in this area because it did not produce these analyses; instead, it relied on an arbitrary range of Pi groups for similarity assessment. As a result, the staff requested GEH to submit additional information to address the concerns identified above.

GDCS Transition Phase

In RAI 6.3-1, the staff requested GEH to perform a revised scaling analysis for the 4500-megawatt-thermal (MWt) ESBWR addressing the deficiencies, as discussed above, including calculating revised Pi values using interconnected volumes and components, and to use the updated ESBWR design values in the analysis.

In response, GEH performed an updated ESBWR scaling analysis in MFN 06-225, "Response to NRC Request for Additional Information Letter No. 4 Related to ESBWR Design Certification Application – ESBWR Scaling Analysis – RAI Number 6.3-1," July 18, 2006 (MFN 06-225). In the updated analyses, GEH used an updated ESBWR power level and design configuration, as well as other design modifications. In addition, GEH addressed the deficiencies in the original scaling approach using new equations that accounted for the interactions between volumes. The system Pi's that resulted from the updated analysis differed significantly from the system Pi's from the noninteracting equations that GEH used in its original analyses.

Moreover, GEH successfully applied the equations to the GDCS transition phase, which is the onset of GDCS injection—the time period when the minimum vessel inventory occurs in a LOCA. In the updated analysis, GEH abandoned the arbitrarily defined range of Pi groups and conducted a rigorous analysis for the GDCS transition phase, where the largest difference in Pi groups was observed. This confirmatory scaling analysis was based on a simplified model for a BWR depressurization transient as documented in M. di Marzo, "A Simplified Model of the BWR Depressurization Transient," Nuclear Engineering and Design, 205 (2001), pgs. 107-114, July 28, 2000. The results showed that, although the variations in Pi groups between the ESBWR and test facilities approached the 1/3 to 3 range, the variations in the pressure and liquid mass responses had a small impact on the figure of merit (minimum RPV liquid inventory, the most critical variable) compared to the margin to the design limit (core uncovery). This confirmed that the experiments behaved qualitatively the same as their scaling model and the TRACG ESBWR model.

Because GEH used the GDCS injection line break in its earlier analyses for the 4000-MWt ESBWR, it also used the same GDCS injection line break for the 4500-MWt ESBWR as an example in the updated analysis to allow a comparison of the results.

GEH also presented the results for the 4500-MWt ESBWR for the base case (with standby liquid control system (SLCS) flow) and another case without the SLCS injection during the late blowdown and the GDCS transition phases. The results showed that the SLCS injection helps keep the water inventory at a higher level until the GDCS injection begins. However, the SLCS injection has only a small effect on the vessel depressurization rate, and thus on the timing of the GDCS initiation which occurs when RPV pressure reaches the pressure at which GDCS injection begins. Also, the calculated values of Pi groups for the late blowdown phase showed that the contribution from the SLCS flow rate is small compared to the ADS flow rate, which dominates the RPV depressurization rate; however, the SLCS flow rate is significant from the viewpoint of RPV liquid inventory. In addition, the ICS and control rod driveline flows were neglected because they are small compared to the break, ADS, SLCS, and GDCS flows.

The GEH results show that the behavior of the 4500-MWt ESBWR during the late blowdown and GDCS injection phases is expected to be very similar to that observed in the GIRAFFE-SIT and GIST tests. Thus, GEH stated that no additional tests were required for scaling of the 4500-MWt ESBWR for these phases.

The staff finds the GEH response acceptable.

Long-Term Cooling Phase

The ESBWR system encompasses two major energy sinks—the SP and the PCCS pool. The SP is the primary sink in the initial portion of the transient. The PCCS pool takes over in the long-term portion of the transient. The transition from heat deposition in the wetwell to heat deposition in the PCCS pool is a fundamental element of the ESBWR system.

GEH scaled and designed the systems test facilities in such a way that little data were obtained regarding multidimensional phenomena. Analysis of the system test data was based on a lumped-parameter approach that eliminated multidimensional spatial variations in the containment. As such, the tests did not provide sufficient data to credit multidimensional effects. Since the data were not suitable to qualify TRACG to predict multidimensional effects, TRACG is not qualified for multidimensional effects in the ESBWR analysis.

In RAI 6.3-1, the staff requested that GEH compare the revised ESBWR Pi values with those obtained from the tests for the LOCA phases.

In response, GEH provided comparisons of revised values of Pi groups between the ESBWR and the tests for all the phases of LOCA. The Pi values were within an acceptable range (1/3 to 3).

GEH did not provide any confirmatory scaling analysis for the phases of LOCA, except for the blowdown and GDCS transition phases. Therefore, in a supplement to RAI 6.3-1, the staff requested that GEH justify why it considered confirmatory scaling analysis similar to the approach taken for the late blowdown and GDCS transition phases to be unnecessary for other phases of LOCA, including the long-term cooling phase.

In response, GEH stated that the purpose of the simplified confirmatory scaling analysis for the late blowdown and GDCS transition phases was to show that the pair of differential equations that govern the RPV transient pressure and liquid inventory could be simplified and solved numerically to directly demonstrate similar responses for the ESBWR and the test facilities. In the process, the key phenomena that govern the relatively rapid changes in the RPV pressure and liquid inventory during these phases of the LOCA transient were identified and clarified. This situation is in marked contrast to the long-term cooling (PCCS) phase of the LOCA transient where pressures in the RPV, drywell, and wetwell are essentially equal, and changes occur in a quasi-static manner.

GEH further stated that the steam generation rate inside the RPV is directly proportional to the decay heat and the entire amount of steam discharges into the drywell through the break and the ADS. The steam discharge rate is independent of the type of break, and the RPV and the drywell are effectively uncoupled. The decay heat steam, along with a small amount of residual drywell noncondensible, flows into the PCCS, which is submerged in the PCCS pool above and outside the containment. The steam is condensed in the PCCS tubes and the condensate flows into the GDCS pool. The residual drywell noncondensible eventually moves to the wetwell gas space and causes a small pressure increase. GEH also stated that, during the long-term cooling phase, as shown in DCD Tier 2, Figure 6.2-11, the PCCS is capable of transferring all the decay heat to the PCCS pool outside the containment. Therefore, there is no further heat up of the wetwell pool and no wetwell gas space pressure increase from steam generated in the RPV because of decay heat. Hence, the only coupling between the drywell/PCCS and the wetwell can be taken into account by considering eventual transfer of all drywell noncondensibles to the wetwell. As a consequence, GEH concluded that, by considering transfer of all drywell noncondensibles to the wetwell, the containment pressure would be conservatively bounding. The staff agrees with this conclusion. However, the staff also believes that potential trapping and delayed release of enough drywell noncondensibles can adversely impact PCCS performance and containment pressure during the long-term cooling phase, and therefore, should be considered when calculating the conservatively bounding containment pressure. In addition, the staff agrees with the GEH statement that the minimal coupling between the different regions during the long-term phase means that the Pi groups for the wetwell and the drywell can be evaluated separately without reference to the other regions.

Based on the considerations that the containment pressures will be calculated on the basis of a bounding approach and that the Pi values are within an acceptable range, GEH concluded that no additional or confirmatory scaling analysis was required for the long-term cooling phase.

The staff finds the GEH response acceptable.

21.5.3.5 Summary of GEH Scaling Analyses

The GEH scaling analyses demonstrated that the test facilities were scaled properly for their intended purpose. All the test facilities met the top-down scaling criteria. However, the power-to-volume scaling approach introduced scaling distortions related to structural heating/cooling, aspect ratio, and geometrical complexity. GENH identified and evaluated these distortions. The staff concluded that the analyses included the essential phenomena that are expected to occur in the ESBWR design and that the experimental results were appropriate for TRACG qualification.

The distortions, as identified by GEH, were caused by heat transfer from RPV structures, heat transfer to and from the drywell and wetwell structures, and drywell three-dimensional effects, including drywell mixing, noncondensible gas stratification, and buoyancy/natural circulation. GEH developed bounding models to address these three-dimensional effects such that TRACG is able to adequately predict the effects. The staff further concluded that the data from the GIRAFFE and the PANDA facilities can be used for scale-up to the ESBWR through the TRACG code. Based on this evaluation, the staff concluded that the TRACG model used for the containment/LOCA evaluation can be conservatively biased.

The staff concluded that GEH demonstrated that relevant and sufficient data exist to qualify TRACG in its simulation of the phase for which the scaling analysis was completed. The phase for which this was done, the GDCS injection phase, is indeed the most important period of the transient. The staff, however, recognizes that there are deficiencies in the GEH scaling analysis for the long-term cooling phase, particularly, regarding the system interactions, the energy partition between the SP and PCCS pools, and the effect of containment structures. However, GEH can employ conservative, bounding analyses for the long-term cooling phase to overcome these deficiencies.

The staff, therefore, finds the GEH scaling analyses acceptable.

21.5.3.6 Compliance with 10 CFR 52.47, “Contents of Applications; Technical Information,” Requirements

The ESBWR met the requirements delineated in 10 CFR 40.43(e) as referenced by 10 CFR 52.47(b)(2)(c)(2), as discussed below:

ESBWR plant features were used in earlier BWR designs which have provided satisfactory operation over a very large number of combined plant operating years of service. While the details of the particular plant feature design for the ESBWR may differ somewhat from those in current plants, the function of each feature is substantially the same. For those ESBWR safety features considered unique, GEH used separate effect test programs to demonstrate their performance. The operating plant experience and the ESBWR specific separate effect test programs constitute a sufficient database to meet the requirements of 10 CFR 50.43(e)(1)(i).

GEH used integral test programs to demonstrate the acceptability of system interactions for features that are unique to ESBWR (i.e. ICS, GDCS, and PCCS). For features that are not unique to ESBWR, operating plant experience is applicable. The operating plant database and the ESBWR integral test program data are sufficient to meet the requirements of 10 CFR 50.43(e)(1)(ii) and (iii).

ESBWR feature performance was predicted with the TRACG computer code. TRACG was qualified by a comparison of data from ESBWR-specific separate effect and integral test programs to operating BWRs over a wide range of reactor conditions, including temperatures and pressures during which the features are expected to operate. The TRACG analyses add to the confidence that the features would perform as expected and that the requirements of 10 CFR 50.43(e) have been met.

21.5.4 Conclusions

The full-size component test data from the PANTHERS/PCCS and PANTHERS/ICS testing programs cover the range of operational conditions expected in the design-basis LOCAs in the ESBWR. These data are adequate for validating the TRACG code regarding the PCCS and ICS performance in the ESBWR, with the understanding that a PCCS condenser in the ESBWR will have approximately 35 percent more heat removal capability than that of the PANTHERS/PCCS condenser, and an ICS condenser (with two identical modules of tubes) has twice the heat removal capability as the PANTHERS/ICS condenser (with only one module of tubes).

The integral systems test data from the GIST, GIRAFFE/Helium, GIRAFFE Systems Interactions, PANDA M-series, and PANDA P-series test programs as a whole cover a range of the late blowdown phase, GDCS phase, and long-term cooling phase of the accidents. Strengths and weaknesses of the individual test programs were identified and evaluated. The staff has reviewed the test programs and results and concludes that the weaknesses (including some phenomenon distortions) in general did not invalidate the overall reactor vessel and containment response in a LOCA simulated by TRACG. The combined data from the GIST, GIRAFFE, and PANDA integral systems tests are generally expected to cover the LOCA phenomena and processes defined in the PIRTs for the late blowdown phase, GDCS phase, and long-term cooling phase.

Furthermore, GEH demonstrated that relevant data are sufficient to qualify TRACG in its simulation of the phase for which the scaling analysis was completed. The phase for which this was done, the GDCS injection phase, is indeed the most important period of the transient. GEH employed conservative, bounding analyses for the remainder of the LOCA event.

On the basis of the above discussion, the staff concludes that GEH has met the requirements of 10 CFR 50.43(e) and that no further testing in support of the LOCA thermal-hydraulic behavior for the ESBWR design is necessary.

21.6 TRACG Analysis Methods for the ESBWR

GEH uses the TRACG thermal-hydraulic code to perform design-basis analyses of the ESBWR. The analysis code and methods for each application are described in the following topical reports:

- Large- and small-break LOCA and containment analysis are described in Section 2 and 3 of NEDC-33083P-A, MFN 05-017, "TRACG Application for ESBWR," March 2005. (ADAMS Accession No. ML051390265) (NEDC-33083P-A)

- Non-LOCA transients, including anticipated operational occurrences (AOOs) and infrequent events (IEs), are described in Section 4, "Transient Analysis," of NEDC-33083P-A.

- Stability analysis is described in NEDC-33083P, Supplement 1, "TRACG Application for ESBWR Stability," December 9, 2004. (ADAMS Accession No. ML0500301602) (NEDC-33083P, Supplement 1)

- ATWS analysis is described in NEDE-33083P, Supplement 2, "TRACG Application for ESBWR Anticipated Transient Without Scram Analyses," January 2006. (ADAMS Accession No. ML060190592) (NEDE-33083P, Supplement 2)

During the preapplication review of the ESBWR, the NRC staff reviewed and approved NEDC-33083P-A for the use of TRACG as an acceptable evaluation model for the LOCA and containment design-bases analyses. NEDC-33083P-A documents staff's evaluation and includes 20 "Confirmatory Items" that were identified as needing resolution at the design certification stage. The staff's Addendum to the Safety Evaluation Report with Open Items for NEDC-33083P-A, Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design (ML073190044) addresses the status of the 20 Confirmatory Items. Five Confirmatory Items are closed. The remaining confirmatory items are associated with the following open items: RAIs 4.8-16, 6.3-46, 6.3-52, 6.3-54, 6.3-55, 6.3-65, 6.3-81, 21.6-55, 21.6-75, 21.6-95, and 21.6-96, 21.6-98, 21.6-103, 21.6-106, 21.6-107, and 21.6-108. Refer to ML073190044 for a detailed discussion of the confirmatory items and associated RAIs.

The NRC staff reviewed and approved NEDC-33083P, Supplement 1, for the use of TRACG as an acceptable evaluation model for the ESBWR stability analysis. NRC Letter dated August 29, 2007, "Reissuance of Safety Evaluation Regarding the Application of the GE-Hitachi Nuclear Energy Americans LLC (GEH) LTR "TRACG Application for the ESBWR Stabilty Analysis," NEDE-33083P, Supplement 1" (ML072270192), documents the staff's evaluation and includes

Confirmatory Items that were identified as needing resolution at the design certification stage. The staff's Addendum to the Safety Evaluation Report (NEDC-33083P, Supplement 1) for TRACG as applied to Stability (ML073190079) addresses the status of the 7 Confirmatory Items. All 7 Confirmatory Items are currently open. The 7 Confirmatory Items are associated with the following open items: RAIs 4.4-59, 4.4-61, 4.8-16, 6.3-54, 6.3-55, 21.6-51 and 21.6-82. Refer to ML073190079 for a detailed discussion of the confirmatory items and associated RAIs.

The NRC staff reviewed NEDC-33083P-A. Section 4, Transient Analysis. The staff's Safety Evaluation Report with Open Items for Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design (ML073100168) addresses the status of the review. The staff's review is associated with the following open items: RAIs 6.3-54, 6.3-55, 21.6-55, 21.6-57, 21.6-61, 21.6-62, 21.6-63, 21.6-64, 21.6-65, 21.6-75, 21.6-78, 21.6-79, 21.6-81, 21.6-84, 21.6-92. Refer to ML073100168 for a detailed discussion of the open items.

The NRC staff reviewed NEDE-33083P, Supplement 2, “TRACG Application for ESBWR Anticipated Transient Without Scram Analyses." The staff's Safety Evaluation with Open Items for Application of the TRACG Computer Code to Anticipated Transients Without Scram for the ESBWR Design NEDE-33083P, Supplement 2," October 2007 (ML073190036) addresses the status of the review. The staff's review is associated with the following open items: RAIs 6.3-54, 6.3-55, 21.6-4, 21.6-8, 21.6-12, 21.6-34, 21.6-35, 21.6-39, 21.6-41, 21.6-44, 21.6-53, 21.6-75, 21.6-83, and 21.6-100. Refer to ML073190036 for a detailed discussion of the open items.

Although the full details of the staff's evaluation, including limitations and conditions of the TRACG code as applied to ESBWR design-basis analyses, can be found in the above references, the following sections document adherence to NRC regulations for the TRACG code for purposes of approving the ESBWR design certification document.

21.6.1 Regulatory Basis

To establish a licensing basis, licensees must analyze transients and accidents in accordance with the requirements of 10 CFR 50.34, “Contents of Construction Permit and Operating License Applications; Technical Information,” 10 CFR 50.46, and where applicable, NUREG-0737, “Clarification of TMI Action Plan Requirements,” issued November 1980.

The staff reviewed the TRACG code based on the review guidelines of the standard review plan (SRP), Section 15.0.2, “Review of Transient Accident and Analysis Methods” U.S. Nuclear Regulatory Commission, December 2005. (ADAMS Accession No. ML053550265) (SRP Section 15.0.2).

21.6.2 Summary of Technical Information

The following sections summarize the technical information needed to evaluate the analysis codes in accordance with the guidance in SRP Section 15.0.2.

21.6.2.1 Documentation

The development of an evaluation model for use in reactor safety licensing calculations requires a substantial amount of documentation. SRP Section 15.0.2 requires that this documentation include/cover (1) the evaluation model, (2) the accident scenario identification process, (3) the code assessment, (4) the uncertainty analysis, (5) a theory manual, (6) a user manual, and (7) the QA program. The following list describes the documentation that GEH provided:

- model description and theory manual (NEDE-32176P, Rev. 3, "TRACG Model Description," April 2006. (ADAMS Accession No. ML061160238) (NEDE-32176P, Rev. 3)

- LTRs that cover accident scenario identification and uncertainty analysis

  — NEDC-33083P-A for LOCA,
  — NEDC-33083P, Supplement 1 for stability,
  — Chapter 4 of NEDC-33083P-A for AOO/IE, and
  — NEDE-33083P, Supplement 2 for ATWS

- code assessment

  — NEDE-32177P, Revision 2, "TRACG Qualification," January 2000. (ADAMS Accession No. ML003683162)

  — MFN 04-059, "Update of ESBWR TRACG Qualification for NEDC-32725P and NEDC-33080P Using the 9-Apr-2004 Program Library Version of TRACG04," June 6, 2004. (ADAMS Accession No. ML041610037)

  — NEDC-32725P, Revision 1, "TRACG Qualification for SBWR," August 30, 2002. (ADAMS Accession Nos. ML022560558 and ML022560559)

- user's manual— UM-0136, Revision 0, "TRACG04A, P User's Manual," December 2005. (ADAMS Accession No. ML071150376)

- QA program

21.6.2.2 Evaluation Model

An evaluation model is the calculation framework for evaluating the behavior of the reactor coolant system during a postulated accident or transient. It includes one or more computer programs and other information necessary to apply the framework to a specific transient or accident, such as mathematical models used, assumptions included in the programs, a procedure for treating the program input and output information, specification of those portions of the analysis not included in the computer programs, values of parameters, and other information necessary to specify the calculation procedure. Evaluation models are sometimes referred to as a licensing methodology.

21.6.2.3 Accident Scenario Identification Process

The accident scenario identification process is a structured process used to identify and rank the reactor component and physical phenomena modeling requirements based on (1) their importance to acceptable modeling of the scenario and (2) their impact on the figures of merit for the calculation. It is also used to identify the key figures of merit or acceptance criteria for the accident.

GEH has performed phenomena identification and ranking and summarized the results in PIRTs. Table 21.6-1 summarizes the PIRTs submitted by GEH for the ESBWR.

Table 21.6-1 ESBWR PIRTs

Scenario	Table	Reference
LOCA—short term (water level calculations)	2.3-1	NEDC-33079P, Revision 1
LOCA—long-term core cooling		MFN 05-105
AOO/IE	2.3-3	NEDC3-3079P, Revision 1
Stability	2.3-5	NEDC-33079P, Revision 1
ATWS	2.3-4	NEDC-33079P, Revision 1

21.6.2.4 Code Assessment

The code assessment provides a complete assessment of all code models against applicable experimental data and/or exact solutions in order to demonstrate that the code is adequate for analyzing the chosen scenario. GEH provided assessment reports of the TRACG code for general and ESBWR-specific qualification in

1. NEDE-32177P, Revision 2, "TRACG Qualification," January 2000. (ADAMS Accession No. ML003683162)

2. MFN 04-059, "Update of ESBWR TRACG Qualification for NEDC-32725P and NEDC-33080P Using the 9-Apr-2004 Program Library Version of TRACG04," June 6, 2004. (ADAMS Accession No. ML041610037)

3. NEDC-3272P, Revision 1, "TRACG Qualification for SBWR," August 30, 2002. ADAMS Accession Nos. ML022560558 and ML022560559).

21.6.2.5 Uncertainty Analysis

Uncertainty analyses are performed to confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest when the code is used in a licensing calculation. The GEH licensing calculations using TRACG are best-estimate methodologies. Table 21.6-2 summarizes the safety parameters calculated by TRACG for the ESBWR.

Table 21.6-2 Safety Parameters Calculated by TRACG

Parameter	Event - Primary (Secondary)
Reactor Vessel Water Level	LOCA (AOO/IE)
Decay Ratio	Stability
Critical Power Ratio	AOO/IE (LOCA, Stability)
Vessel Pressure	AOO/IE, ATWS

Peak Cladding Temperature	ATWS
Suppression Pool Temperature	ATWS

GEH does not explicitly calculate the uncertainty in the reactor vessel water level for LOCA evaluations. Since all of the TRACG LOCA evaluations show that the core does not uncover during a LOCA, GEH instead performed the calculation using bounding assumptions.

Although different methods are used to evaluate the uncertainty for AOO/IE, ATWS, and stability, the uncertainty in the calculated safety parameter is evaluated by statistically combining the uncertainties for medium and/or highly ranked PIRT parameters. In addition to the statistically evaluated uncertainty, extra uncertainty is added in the decay ratio calculation for predicting stability margins by setting the acceptance criteria for the decay ratio at 0.8. An unstable condition would occur at a decay ratio of 1.0. By setting the acceptance criteria at 0.8, GEH allows for an additional uncertainty of 0.2 in the decay ratio calculations.

21.6.2.6 Quality Assurance Plan

The code must be maintained under a QA program that meets the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. Staff performed two audits of GEH's QA plan. The first audit took place between October 16 and October 19, 2006, resuming for the period between October 30 and November 3, 2006 (October 2006 Audit). The second audit took place between December 11 and December 15, 2006, resuming for the period between December 19 and December 20, 2006 (December 2006 audit).

GEH has procedures that meet with the requirements of Appendix B to 10 CFR Part 50 for assuring the quality of its engineering computer programs (ECPs). These procedures specify such things as the types of documentation necessary, control of the change process, and the approval process for ECPs. A synopsis of these procedures follows.

GEH refers to an ECP that is approved for development as "Level 1." A "Level 2" ECP is an approved production program that is verified and documented for design applications or for technical activities used in developing design-related information. The Level 2 review process consists of two phases. In Phase 1, a review team determines the adequacy of the ECP models and specifications and the adequacy of the planned testing. In Phase 2, a review team performs a technical review and provides the final independent verification of the testing. As part of this review phase, GEH ensures that all open items are closed, confirms that the documentation is sufficient and complete, and performs licensing impact evaluations (for NRC-approved methodologies).

GEH uploads Level 2 codes into the program library which can then be used by the GEH staff for the stated design applications. These codes cannot be changed once they have attained Level 2 status. If GEH were to change the code or make error corrections, the code would no longer be considered a Level 2 ECP.

Under certain circumstances, GEH uses non-Level 2 ECPs for design tasks for a limited time. These ECPs are designated as "Level 2R." Although there are no specific QA procedures for a Level 2R code, these codes are typically very close to being a Level 2 code; either they are in the review process or they have had minor changes or error corrections and do not fully satisfy the Level 2 requirements. The QA approving official must still approve Level 2R codes.

At the time NRC approved TRACG04 for application to ESBWR LOCA analysis (NEDC-33083P-A), GEH considered TRACG04 to be a Level 1 ECP. During the December 2006 audit of TRACG at GEH, the staff viewed documentation associated with the Level 2 review process for TRACG04. TRACG04A (the "A" designator refers to the Alpha VMS version) obtained Level 2R status on July 29, 2005, and Level 2 status on August 2, 2005. GEH uploaded the TRACG04A version that is Level 2 and is also used for ESBWR design calculations into the GEH program library on June 27, 2005. This corresponds to Version 52 of TRACG04A.

During the December 2006 audit, the audit team found that GEH controlled changes to TRACG04P PL 52 code under a level 2R code change control process to support code development that did not have QA controls for independent verification and validation (V&V) of code calculations. Considering planned model revisions disclosed during the audit, the final ESBWR DCD revision will likely be based on a later TRACG code revision that the staff has not reviewed. After changes to the TRACG04P code are complete, GEH is required to place the TRACG04 code under a QA approved code change control process (such as Level 2) where independent V&V is performed in accordance with 10 CFR Part 50, Appendix B, Criterion III, Design Control. In RAI 21.6-109, staff requests that GEH Inform the staff when GEH places the TRACG04 code under a QA approved code change control process and provide information to the staff sufficient for the staff to review and approve the version of TRACG04 used to develop the final ESBWR DCD submittal. **RAI 21.6-109 is being tracked as an open item.**

During the review of TRACG for ESBWR LOCA applications, GEH transitioned from TRACG02A to TRACG04. To support this review, GEH submitted the ESBWR-specific qualification cases using TRACG04 (MFN 04-059, "Update of ESBWR TRACG Qualification for NEDC-32725P and NEDC- 33080P Using the 9-Apr-2004 Program Library Version of TRACG04," June 6, 2004. (ADAMS Accession No. ML041610037)). In this submittal, GEH updated NEDC-32725P, "TRACG Qualification for SBWR," and NEDC-33080P, "TRACG Qualification for ESBWR," by combining them into one document and performing the assessment cases using TRACG04. Most of the cases were run with the April 9, 2004, program library version of TRACG04A. This corresponds to Version 40. The team viewed all of the changes from Version 40 up to Version 52 (i.e., the Level 2 version used for all ESBWR design certification calculations).

Although GEH performed some of the TRACG04 assessment with a different version of the code than what it is using to license the ESBWR, the staff determined that the nature of the changes would not invalidate the qualification basis used to support the NRC's approval in NEDC-33083P-A, and therefore finds the changes made to the TRACG04 code acceptable.

The staff found that GEH submitted a code for NRC approval that had not completed the QA process (i.e., Level 1 or Level 2A). However, the staff verified that the executable used for TRACG04A had not been changed since June 27, 2005, and the QA was completed in August 2005. All ESBWR design certification analyses have been performed with this same TRACG04 source code. The staff found that the changes were insignificant overall and that the use of TRACG04 for ESBWR licensing applications complies with the intent of Step 4 of the code, scaling, applicability, and uncertainty methodology, "Frozen Code Version Selection" (NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident," December 1989 (ADAMS Accession No. ML070310119)). The staff found that the

GEH QA procedure for a code to attain Level 2 status is rigorous and meets the requirements of 10 CFR Part 50, Appendix B, Criterion III, Design Control.

In RAI 21.6-92, the staff requested that GEH provide more detailed information on the exact code revision and version numbers used for all LOCA, stability, AOO, IE, and ATWS events. **RAI 21.6-92 is being tracked as an open item.**

21.6.3 Staff Evaluation

The following sections document the basis for the staff's approval of the technical information submitted by GEH for the TRACG code in accordance with the guidance in SRP Section 15.0.2. GEH uses the TRACG coupled thermal-hydraulic and neutronic code to analyze the following DBAs:

- large- and small-break LOCA and containment analysis
- non-LOCA transients, including AOOs and IEs
- stability analysis
- ATWS analysis

The below references fully document the staff's review of the LTRs for the above applications. The information is repeated and consolidated here for convenience to the reader. Some of the bases for the staff's acceptance of these licensing methodologies is proprietary and will not be discussed in detail in this document; however, these bases are documented in the below mentioned references:

- NEDC-33083P-A, MFN 05-017, "TRACG Application for ESBWR," March 2005. (ADAMS Accession No. ML051390265)

- 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. ML072270267 and ML072270276)

- Addendum to the Safety Evaluation Report with Open Items for NEDC-33083P-A, Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design (ML073190044)

- Addendum to the Safety Evaluation Report (NEDC-33083P, Supplement 1) for TRACG as applied to Stability (ML073190079)

- Safety Evaluation Report with Open Items for Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design (ML073100168)

- Safety Evaluation with Open Items for Application of the TRACG Computer Code to Anticipated Transients Without Scram for the ESBWR Design NEDE-33083P, Supplement 2," (ML073190036)

21.6.3.1 Documentation

The staff reviewed the documentation submitted by GEH. The staff determined that GEH included all of the documentation that describes (1) the evaluation model, (2) the accident scenario identification process, (3) the code assessment, (4) the uncertainty analysis, (5) a theory manual, (6) a user manual, and (7) the QA program.

The TRACG LTRs for LOCA (NEDC-33083P-A), AOO/IE (NEDC-33083P-A), stability (NEDC-33083P, Supplement 1,), and ATWS (NEDE-33083P, Supplement 2) provide an overview of the respective evaluation models that describes all parts of the evaluation model, the relationships between them, and where they are located in the documentation. These topical reports also describe the accident scenario including plant initial conditions, the initiating event, and phases of the accident. The topical reports include documentation on the important physical phenomena, systems, and component interactions that influence the outcome of the accident. (NEDC-33083P-A does not include any information about the long-term core cooling phase of the LOCA. The staff received this information in MFN 05-105, Hinds, D.H., General Electric, Letter to U.S. Nuclear Regulatory Commission, “TRACG LOCA SER Confirmatory Items (TAC # MC868), Enclosure 2, Reactor Pressure Vessel (RPV) Level Response for the Long Term PCCS Period, Phenomena Identification and Ranking Table, and Major Design Changes from Pre-Application Review Design to DCD Design,” October 6, 2005, (ADAMS Accession No. ML053140223).

The topical reports also contain a determination of the code uncertainty for a sample plant calculation. NEDC-33083P-A demonstrates the bounding LOCA calculation since GEH did not determine an uncertainty for this event.

In NEDE-32177P, Revision 2,,MFN 04-059, and NEDC-32725P, Revision 1, GEH provided the code assessment for TRACG. These documents comprise a description of each assessment test, why it was chosen, success criteria, diagrams of the test facility that show the location of instrumentation that was used in the assessment, a code model nodalization diagram, and all code options used in the calculation. RAI 21.6-75 requested GEH to provide an update to the TRACG qualification report (NEDE-32177P, Rev. 2) that is consistent with the current version of TRACG used in the ESBWR licensing analyses (TRACG04). In response, GEH submitted Revision 3 of the TRACG qualification report on August 29, 2007. **RAI 21.6-75 is being tracked as an open item pending staff review of NEDE-32177P, Revision 3.**

The staff determined that NEDE-32176P, Revision 3, “TRACG Model Description,” April 2006, is a self-contained document that describes the field equations, closure relationships, numerical solution techniques, and simplifications and approximations (including limitations) inherent in the field equations and numerical methods and limits of applicability for all models in the code.

The staff determined that the TRACG user manual (UM-0136, Rev. 0, “TRACG04A, P User’s Manual,” December 2005) provides detailed instructions about how the computer code is used; a description of how to choose model input parameters and appropriate code options; guidance about code limitations and options that should be avoided for particular accidents, components, or reactor types; and, if multiple computer codes are used, documented procedures for ensuring complete and accurate transfer of information between different elements of the evaluation model. The LTRs (NEDC-33083P-A, NEDC-33083P, Supplement 1, and NEDE-33083P, Supplement 2) provide additional guidance on specific modeling of the events.

During the December 2006 audit of GEH records, the staff reviewed the GEH documentation for a QA plan that describes the procedures and controls under which the code was developed and assessed, as well as the corrective action procedures that are followed when an error is discovered.

The staff requested that GEH update its documentation to reflect the current status of the code and current ESBWR plant design applicability. NEDC-33083P-A gives the application methodology and is based on the preapplication (4000-MWt) design and TRACG nodalization. RAI 21.6-98 requests that GEH describe all design changes since the approval of TRACG for ESBWR LOCA analyses in NEDC-33083P-A and demonstrate that the staff's conclusions would not be altered as a result of these changes. RAI 21.6-63 requests that GEH provide the differences between what is described in NEDC-33083P-A and NEDE-33038P, MFN 04-109, Section 4.7, "Demonstration Calculations for ESBWR AOOs," October 8, 2004, and the current application methodology used in the ESBWR DCD. **RAIs 21.6-98 and 21.6-63 are being tracked as open items.**

Although GEH is providing some of the appropriate updates to the documentation in RAI responses, in RAI 21.6-63, Supplement No. 1, and RAI 21.6-65, Supplement No. 2, the staff requested that GEH submit these updates in a single consolidated document. Specifically, the staff requested that GEH submit an update to the AOO portion of the TRACG topical report (Chapter 4 of NEDC-33083P-A) as either a stabd alone new topical report or a new supplement to NEDC-33083. **RAI 21.6-63 and RAI 21.6-65 are being tracked as open items.**

21.6.3.2 Evaluation Model

TRACG employs a two-fluid model for two-phase flow. It solves six conservation equations for both the liquid and gas phases, along with phasic constitutive relations for closure. In addition, a boron transport equation and a noncondensible gas mass equation are solved. The spatially discretized equations are solved by donor-cell differencing in staggered meshes in one, two, or three dimensions. TRACG is used for both reactor vessel and containment. The list of constitutive models covers all important phenomena that may occur in a BWR, SBWR, or ESBWR.

21.6.3.2.1 Counter-Current Flow Condition

The action of steam flowing upward can impede the downward flow of cooling water and lead to the counter-current flow condition. GEH assessed the TRACG counter-current flow limitation (CCFL) model with data from the CSHT test facility. Comparisons documented in NEDE-2177P for liquid temperatures near saturation versus TRACG demonstrate that the code provides excellent agreement for saturated liquid. Agreement with subcooled liquid is excellent with steamflow rates that are less than the condensation capacity. For flow rates greater than the condensation capacity, the average deviation between liquid downflow predicted by TRACG is within the measurement error. Accordingly, the staff concludes that TRACG adequately predicts saturated CCFL and subcooled CCFL breakdown.

21.6.3.2.2 Heat Conduction

TRACG solves the heat conduction equation for the fuel rods (in cylindrical geometry) and for structural materials (in slab geometry) in the system. The latter has either a lumped slab model or a one-dimensional slab model. The strengths of the TRACG heat conduction model are the sophisticated transient gap conductance model and the implicit solution method that couples the heat transfer between the fuel rod and the coolant by iteration. The staff concludes that TRACG appropriately provides for solution of heat conduction.

21.6.3.2.3 Wall Heat Transfer

TRACG has a very detailed wall heat transfer model based on the boiling curve. The model has standard heat transfer regimes—single-phase liquid or vapor, nucleate boiling, critical heat flux (CHF), transition boiling, film boiling, and condensation with and without the effect of noncondensibles. There are correlations for transitions between different heat transfer regimes. The correlations for different regimes are standard correlations from the literature.

The code has been assessed with a variety of tests that have become the standards for assessing wall heat transfer. The assessments included thermal-hydraulic test facility (THTF) tests for film boiling heat transfer, CSHT tests that included thermal radiation heat transfer, and THTF tests for boiling transition, as well as critical power data gathered at the ATLAS facility. The staff concludes that the breadth and accuracy of the assessment cases demonstrates acceptable capability for TRACG to predict wall heat transfer.

21.6.3.2.4 Postcritical Heat Flux Heat Transfer

TRACG has a rewet model for post-CHF heat transfer. With the exception of ATWS, none of the TRACG applications for ESBWR experience post-CHF heat transfer. The NRC staff is still reviewing the applicability of this model for ESBWR ATWS events as discussed in Safety Evaluation with Open Items for Application of the TRACG Computer Code to Anticipated Transients Without Scram for the ESBWR Design NEDE-33083P, Supplement 2," ML073190036.

21.6.3.2.5 Flow Regime Maps

A two-fluid formulation relies upon models for estimating interfacial transfer rates for mass, momentum, and energy. The models for interfacial processes, in turn, rely on the shape and size of the interface. Common practice is to develop flow regime maps to identify the distinct regime for two-phase distribution. The knowledge of the flow regime allows the code to select applicable correlations for transport processes.

The flow regime maps are generally two-dimensional maps between void fraction and mass flux. TRACG used this approach to identify the two-phase flow regimes. It also has correlations for entrainment for dispersed flow regimes. Transition between annular flow and dispersed droplet flow is given by the onset of entrainment. For low vapor flow, annular flow will exist, and, as the vapor flux is increased, more and more entrainment will occur, causing a gradual transition to droplet flow.

The models for flow regime transitions in TRACG04 are qualified at low and high pressure. As documented in NEDC-33083P-A and in staff's Addendum to the Safety Evaluation Report with Open Items for NEDC-33083P-A, Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design (ML073190044), the staff reviewed the flow regime maps and transition between flow regimes and found them acceptable for the stated ESBWR applications.

21.6.3.2.6 Interfacial Shear

The interfacial shear model was derived from the drift flux model using available experimental data at steady state. The models are based on current state-of-the-art technology and have been assessed with a large database covering the range of conditions that are expected in the reactor. The code uses a critical Weber number criterion for estimating interfacial area density or bubble/droplet diameter. However, there are differences in the way this approach is used for interfacial momentum and heat transfer in bubbly flow and droplet flow. NEDE-32176P, Revision 2, provides an assessment of the interfacial shear through the capability of TRACG to predict void fraction data including single tube data, rod bundle data, and data for large hydraulic diameters. The test conditions used in assessment cover both adiabatic tests, in which there is no effect of heat transfer on the void fraction, and heated tests. The tests cover a wide range of flow conditions with pressure, flow rate, and inlet subcooling varied. Comparisons between TRACG and test data from sources such as the FRIGG and Christensen tests show calculations to be within the measurement error for the tests. The staff concludes that this demonstrates acceptable capability to predict interfacial shear.

The drift velocity used to calculate interfacial shear in the dispersed annular flow regime is based on the entrainment fraction. The staff will review the GEH qualification of its void fraction data provided in Revision 3 of the qualification report to ensure that the modifications to the entrainment fraction and its subsequent use in the interfacial shear model compare well with data. The staff requested in RAI 21.6-75 that GEH submit the updated qualification report. In response, GEH submitted Revision 3 of the TRACG qualification report on August 29, 2007. **RAI 21.6-75 is being tracked as an open item pending staff review of NEDE-32177P, Revision 3.**

21.6.3.2.7 Wall Friction and Form Losses

The wall friction and form losses are important for predicting single- and two-phase flows. The code has standard models consisting of Moody curves for single-phase flow and a two-phase multiplier based on the Chisholm correlation. Similarly, there is a standard model for form losses for abrupt area changes. The staff recognizes that simplifying assumptions are often necessary or expedient in computer code simulation of two-phase flow phenomena. However, the induced errors caused by simplifying assumptions should be understood. GEH determined those errors in the assumption of consistent wall friction and form loss partitioning between phases through code assessments using data from the FRIGG, Christensen, Wilson, and Bartolomei test programs (NEDE-32176P, Rev. 2). In all assessment cases, the prediction-measurement standard deviation was shown to be on the order of the measurement error. In addition, wall friction assessments have been performed using data from the ATLAS facility over a range of flow conditions. The prediction-measurement comparisons show a calculated error rate on the order of the measurement error. The staff concludes that these assessments demonstrate acceptable capability to predict wall friction and form loss.

21.6.3.2.8 Critical Flow

Critical flow is calculated using coarse-mesh nodalization and semiempirical approximation for choking criteria. The critical flow model also allows for choking in the presence of noncondensible gases. The critical flow model in TRACG has been assessed against data from the Marviken critical flow tests, pressure suppression test facility (PSTF), and Edwards test. The Edwards and PSTF tests are small-scale tests, and the Marviken tests are large scale. In each blowdown period, the measured and predicted mass flows were in good agreement with the predicted bounding the measured. Timing of the transition was also in good agreement. The predicted mass flow rates were generally conservative compared with the data in the smaller scale tests. Comparison of TRACG predictions versus data from tests in different scale test facilities show that TRACG generally overpredicts the data and is therefore conservative. The critical flow model is detailed, well defined, and acceptable for predicting choked flow.

There is an open item related to the choked flow model used in ESBWR LOCA analyses. In RAI 6.3-13, the staff asked GEH to include the RPV injection line nozzle and equalizing line nozzle throat lengths in inspection, test, analysis, and acceptance criteria to ensure that the ratio of the length to diameter remains within the applicability range of the TRACG code flow choking model for LOCA calculations. Section 6.3 of NEDE-32176P, Revision 3, describes the TRACG04 choked flow model. Section 6.3.3 describes the calculation of the sonic velocity. In this section (page 6-51), GEH states the simplifying assumptions used to calculate the sonic velocity. Under this list, GEH states in Section 6.3.3.1, page 6-51, that it assumes equilibrium conditions. GEH states that, “Under certain circumstances, the equilibrium assumption may break down. In particular, for break assemblies of very short length, non-equilibrium transport behavior may be important.” In a supplemental RAI, the staff asked that GEH address questions related to the applicability of the TRACG04 flow choking model to the ESBWR RPV injection line and equalizing line nozzles. **RAI 6.3-13 is being tracked as an open item.**

21.6.3.2.9 Two-Phase Level Tracking

A two-phase level may exist in the bypass, lower plenum, downcomer, chimney, drywell, and wetwell. The two-phase level-tracking model invokes some approximations for the void fraction above and below the mixture level that may not be accurate if significant voiding occurs below the mixture level. The model has been assessed with PSTF-level swell tests. Comparisons of predicted versus measured level indicate that TRACG was generally able to predict the measured level to an accuracy consistent with the measurement uncertainty. Sensitivity studies were also performed on nodalization, convergence ratio, and time step size. Little sensitivity was found in the studies. The staff concludes that TRACG adequately models level swell as evidenced by the code predictions falling within the experimental measurement uncertainty.

21.6.3.2.10 Flashing

Vapor generation or flashing is an important phenomenon for any depressurization transient such as a LOCA. The vapor generation is predicted by energy balance at the interface, where the differences in heat fluxes result in phase change. TRACG has a mechanistic model for interfacial heat transfer that depends upon interfacial area and the shape of the interface. The interface is defined on the basis of flow regime. The model has been assessed with a variety of tests. TRACG predictions were reasonable, indicating that the models are applicable to LOCAs. The staff notes that there are limits on bubble size and number density in bubbly and droplet flow regimes. The good comparison to the void fraction and heat transfer data shows that these limits do not adversely affect the results. As is the case with all thermal-hydraulic system codes today, there is an inconsistency in interfacial area used for momentum transfer and heat transfer in bubbly and droplet flows.

21.6.3.2.11 Minimum Stable Film Boiling Temperature

For the minimum stable film boiling temperature, GEH used the Iloeje correlation for ESBWR applications. TRACG has the option of using the Shumway correlation, and GEH stated that the flow and pressure dependence are captured better for the Shumway correlation. The staff has not reviewed the Shumway correlation and finds the use of the Iloeje correlation acceptable for ESBWR applications for LOCA, AOO, and ATWS. For LOCA and AOO events, the core does not enter film boiling and therefore this correlation is not used. For ATWS events in which the core does go into film boiling, the minimum stable film boiling temperature is only used to determine when the core will quench. The staff asked, in a supplemental question to RAI 21.6-79, that GEH justify why this parameter was ranked high for the TRACG application for BWR/2-6 AOOs (NEDE-32906P-A, Revision 2, MFN 06-046, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis," February 28, 2006.) **RAI 21.6-79 is being tracked as an open item.**

21.6.3.2.12 Critical Heat Flux

For CHF, TRACG has a proprietary correlation, the General Electric critical quality boiling length correlation (GEXL), based on the critical quality concept for normal flows, and uses a modified Zuber correlation for low flows and flow reversal. The NRC approved the GEXL correlation for specific fuel designs. Section 4.4 of this report discusses the use of the GEXL correlation for the ESBWR. In response to staff RAI 4.4-31, Supplement 1, GEH submitted data showing that the TRACG time-to-boiling transition calculation may be nonconservative. The staff requested in RAI 21.6-104 that GEH justify that this calculation is either conservative or bounded by the code uncertainty. GEH also shows critical power ratio (CPR) plots during low-power, low-pressure conditions (startup, LOCA). Since these conditions are outside the applicability range of the GEXL correlation, in RAI 4.4-61, the staff requested additional information on how these values were calculated. **RAIs 4.4-31, 4.4-61 and 21.6-104 are being tracked as open items.**

21.6.3.2.13 Gap Conductance

TRACG has the option for using either a constant or dynamic gap conductance. The dynamic gap conductance is used in ESBWR for AOO/IE, stability, and ATWS events. For LOCA events GEH chose the constant gap conductance option. These gap conductance values are generated using the GESTR thermal mechanical code.

The dynamic gap conductance is modeled as the sum of the contributions from radiation heat transfer, thermal conduction through the gas mixture in the asymmetric radial gap, and conduction through the fuel/cladding contact spots. The conductance of the gas in the radial gap depends on the effective gap size and accounts for the asymmetric radial displacement of cracked pellet wedges. The contact conductance depends on the size of the gap after accounting for fuel and cladding thermal expansion. The radiation heat transfer between the fuel pellet and cladding is modeled by a conventional radiation heat transfer coefficient with separate thermal emissivities for the pellet and clad surface.

The conductance across the gas gap is calculated as the gas gap thermal conductivity divided by an effective gap modified for temperature jump at the gas-solid interface and the effect of discontinuous gas gap resulting from contact spots. The gas in the gap is composed of helium and fission gas. An effective thermal conductivity is calculated for the gap gas. The helium pressure, composition of the fission gas, and relative amount of xenon and krypton in the fission gas are all obtained from the GESTR fuel files.

The internal gas pressure is calculated by considering gas in the volume along the length of the fuel column and the gas in the fuel rod plenum. Outputs from GESTR-LOCA are used to calculate the initial fuel column volume-to-temperature ratio. The staff concluded in Reference 0 that this use of the NRC-approved GESTR-LOCA method to set initial steady-state conditions is acceptable. In addition, the resulting cladding hoop stress is conservatively predicted.

The cladding average temperature at the maximum linear heat generation rate axial position is used in calculating the growth in the volume of the fuel gas plenum in a transient from thermal expansion. The plenum gas temperature is calculated separately from the gas temperature in the gap of the fuel column.

TRACG has models for gap conductance after cladding perforation. The gas conductivity is adjusted to reflect the presence of a stochiometric mixture of steam and hydrogen from metal-water reaction. The constants in the equation for the gas conductivity in a perforated fuel rod are TRACG input constants. ESBWR is not expected to experience rod perforation during any LOCA, stability, AOO/IE, and ATWS event. However, the staff did review this model, which is documented in NEDC-33083P-A. The TRACG gap perforation model is comparable to the model in SAFER, a code previously reviewed and approved by the NRC. The cladding rupture stress and plastic strain are based on experimental data that the staff has reviewed and approved.

GEH demonstrated that the calculated transient gap responses are in good agreement with those calculated by SAFER/GESTR. The staff's SER included in NEDC-33083P-A documents additional details on the staff's review of the TRACG gap conductance model.

21.6.3.2.14 Fuel Rod Thermal Conductivity

The TRACG04 default model for fuel rod thermal conductivity is based upon the PRIME03 thermal mechanical code. The PRIME03 model accounts for exposure and gadolinium content; the GSTRM model (used in previous TRACG codes) does not. TRACG has options to use either the GSTRM thermal conductivity model or the PRIME03 model. For the LOCA, stability, AOO, IE, and ATWS events analyzed for ESBWR, GEH used the PRIME03 thermal conductivity model in TRACG04. The NRC staff has not reviewed and approved the PRIME03 thermal mechanical fuel code. The staff asked GEH in RAI 6.3-54 to justify use of this model. In response, GEH states that it used GSTRM to generate the gap conductance values employed in the ESBWR LOCA analyses. The staff finds the use of the PRIME for fuel thermal conductivity and GSTRM for gap conductance inconsistent. In RAI 6.3-55, the staff asked GEH to justify this inconsistency. The staff has concerns about the use of GSTRM as identified in the GEH response to RAI 4.8-16. This issue will be resolved in conjunction with the resolution of RAI 4.8-16. **RAI 6.3-54, 6.3-55 and RAI 4.8-16 are being tracked as open items**.

21.6.3.2.15 Distribution of Channel Power

The staff has questions related to the distribution of the channel power as calculated by TRACG, which were communicated to the applicant in RAI 21.6-81. **RAI 21.6-81 is being tracked as an open item.**

21.6.3.2.16 Neutron Kinetics Model

TRACG can perform three-dimensional neutron kinetics calculations. To perform these calculations, TRACG uses input from the PANAC11/TGBLA06 codes. The staff reviewed the PANAC11 and TGBLA06 methods in detail. Section 4.3.3 of this report discusses the staff's review. This section briefly discusses TRACG-specific models and the interface between TRACG and PANAC11/TGBLA.

TRACG04 has a one group, coarse-mesh, nodal diffusion model with six delayed neutron precursor groups. The nodal flux calculation is the same as that performed in the PANAC11 BWR core simulator. The transient flux solution is obtained by integrating the differential neutron precursor and flux equations over space and time and solving the equations by employing a discontinuous flux and continuous current approximation. TRACG also uses PANAC11/TGBLA06 generated cross sections as input by means of a PANAC11 "wrapup" file. GEH submitted the contents of the wrapup file. The staff reviewed this to determine that the information transmitted to TRACG adequately represents the nuclear cross sections. The staff also reviewed and found acceptable the initialization of the TRACG steady-state power distribution to that from PANAC11, given that the two codes have different thermal-hydraulic models. Staff SERs for "ESBWR Transient Analysis" of Safety Evaluation Report for GE14 for ESBWR Nuclear Design Report (NEDC-33239P) and Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring (NEDE-33197P) will include the staff's review of these processes. The review of these topical reports is ongoing.

The PANAC11 void fraction model is based on the Findlay-Dix correlation. The staff has questions regarding the applicability of this correlation to the ESBWR. The staff requested additional information on the uncertainty and applicability associated with the correlation and how it is incorporated into the ΔCPR calculation performed using TRACG and ultimately the operating limit maximum CPR limit. This is an open item that will be addressed by the staff in Section 4.4 of this report. Resolution of this open item may impact this portion of the review.

The important neutronics parameters for ESBWR AOOs are void coefficient reactivity feedback, Doppler reactivity feedback, scram reactivity, and three-dimensional kinetics.

21.6.3.2.16.1 Void Coeffiecient

The void coefficient determines the power spike given a void collapse that results from a pressurization event or cold water event. The void coefficient is implied in how the TRACG neutronics parameters (i.e., infinite multiplication factor, migration area, fast group removal cross section, and fast group diffusion coefficient) change as the local void fraction or moderator density changes. The dominant neutronics parameter for changes in void coefficient is the infinite multiplication factor. The uncertainty and biases for the infinite multiplication factor, which is a function of history weighted moderator density and exposure, is determined by comparing TGBLA06 (the GE lattice physics code) and MCNP. In RAI 21.6-84, the staff asked GEH if it updated the TGBLA comparisons with MCNP for TGBLA06 because, the last time this was reviewed by the staff, GEH used comparisons between MCNP and TGBLA04. GEH provided the reference for the documentation in which this evaluation was performed. The NRC staff is currently reviewing the referenced document. The reference did not describe the lattices used to perform the evaluation. The staff asked in a supplement to RAI 21.6-84 that GEH provide the lattices used to determine if they are applicable for the ESBWR. **RAI 21.6-84 is being tracked as an open item.**

In MFN 05-133, Stramback, G., General Electric, Letter to U.S. Nuclear Regulatory Commission, “Response to DSS-CD TRACG LTR RAIs,” November 11, 2005, (MFN 05-133) GEH also attempts to address the uncertainty associated with using MCNP as the basis for determining the nuclear parameter uncertainty as compared to TGBLA, which is used to generate the neutronics parameter fits used by TRACG. The Monte Carlo method used by MCNP has an uncertainty associated with the number of histories necessary to calculate the infinite multiplication factor for a given fuel design and for a given set of isotopic concentrations (i.e., given burnup or exposure). GEH indicates that for 2 million histories this uncertainty is typically small compared to the uncertainty associated with comparing TGBLA to MCNP. The other major uncertainty to this process is the uncertainty associated with the isotopic concentrations for a given exposure and history weighted moderator density. In MFN 05-133, GEH indicates that this uncertainty is covered by performing TRACG calculations at beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) for an equilibrium core. It is not obvious that this bounds the uncertainty associated with isotopic concentrations. For example, at BOC the fresh fuel assemblies loaded into the reactor have some uncertainty in the uranium-235 and burnable poison concentrations as a result of manufacturing tolerances. The burned fuel loaded into the equilibrium core will have some uncertainty associated with the burn/exposure calculations required to define the equilibrium core. It is not obvious how this uncertainty at BOC will be bounded by performing calculations at MOC and EOC. A range of exposures will be analyzed, but that range continues to have a set of uncertainties that the TRACG uncertainty analysis does not appear to explicitly include. In addition, all uncertainties are included into the infinite multiplication factor, and no uncertainties are included in the migration area, fast group removal cross section, and fast group diffusion coefficient. The migration area, fast group removal cross section, and fast group diffusion coefficient are assumed to be a function of weighted moderator density. Since the weighted moderator density includes uncertainty associated with the GEH void fraction correlation, some uncertainty will be associated with these nuclear parameters, even if these parameters are not a function of exposure or isotopic concentrations. The NRC staff and its contractors (i.e., Brookhaven National Laboratory) are performing independent sensitivity studies using MCNP and MONTEBURNS to evaluate the GEH methodology for calculating void coefficient uncertainty and the isotopic concentration uncertainty. This is an open item and will be addressed in Section 4.3 of this report.

Safety Evaluation Report with Open Items for Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design (ML073100168) provides more details on the staff’s review of the void coefficient bias and uncertainty.

21.6.3.2.16.2 Doppler Coefficient

The Doppler coefficient is a function of exposure and moderator density and has an uncertainty of 4 percent (NEDC-33083P-A). The Doppler coefficient simulates the resonance absorption in uranium and plutonium and the broadening of the resonance absorption as the fuel temperature increases. Therefore it is negative. In response to RAI 21.6-60, GEH provided the results of sensitivity studies that were performed by perturbing the Doppler uncertainty and showed that there was little sensitivity to ΔCPR/ICPR for the loss of feedwater heating (LFWH) and the generator load rejection with a single failure in the turbine bypass system. GEH also showed that there was little sensitivity to the peak pressure to the main steam isolation valve (MSIV) closure event. In addition, although GEH ranks this parameter as having “medium” importance, it still includes the uncertainty in the uncertainty analysis for ESBWR AOO and IE. The staff recognizes that the uncertainty for the Doppler coefficient was determined based on PANAC10. During the review of TRACG04 as applied to the operating fleet (NEDE-32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," May 2006. (NEDE-32906P, Supplement 3)) the staff requested that GEH justify that the uncertainty is applicable or bounding for PANAC11. Similar information has been requested in RAI 21.6-108 in regard to Topical Report NEDC-33083P-A, "TRACG Application for ESBWR". **RAI 21.6-108 is being tracked as an Open Item.**

21.6.3.2.16.3 Scram Reactivity

TRACG simulates the scram reactivity by changing the neutronics parameters from uncontrolled to controlled as the control rods move into the core. The timing of the control rod movement is determined by user input and by trips relative to specified control points in the TRACG model (e.g., turbine control valve closing, MSIV closing). The worth associated with the control movement is determined by how the neutronics parameters change from uncontrolled to controlled (i.e., control rod is present or not present in the core bypass next to the fuel assemblies simulated).

The staff requested that GEH justify the uncertainty values chosen for the scram reactivity as part of the review for migration to TRACG04/PANAC11 for BWR/2-6 AOOs (NEDE-32906P, Supplement 3). These uncertainties were established using the PANAC10 model and are also being used for the TRACG04/PANAC11 application for ESBWR AOO and IE. GEH will provide the justification for these uncertainties in response to staff RAIs related to NEDE-32906P, Supplement 3, Similar information has been requested in RAI 21.6-108 in regard to Topical Report NEDC-33083P-A, "TRACG Application for ESBWR". **RAI 21.6-108 is being tracked as an Open Item.**

21.6.3.2.16.4 Boron Reactivity

TRACG04 models the negative reactivity from boron by adjusting the absorption cross section for other preexisting neutron removal mechanisms already modeled in TRACG.

The staff believes that GEH accounts for all of the factors affecting boron reactivity in the development and testing of its empirical model. However the only validation available for staff review is a code-to-code comparison to PANAC11. The staff is performing independent MCNP calculations to compare the efficacy of the empirical cross-section model to account for the spectral shift consistent with MCNP calculations performed for different fuel lattices at a variety of exposure histories to examine the impact of exposure effects on the instantaneous nodal thermal spectrum. In addition, GEH determined the uncertainty of the model using code-to-code comparisons with TGBLA06/PANAC11. The staff's independent calculations will also assist in determining if the 1-percent uncertainty is appropriate. The staff calculations are ongoing. This is an Open Item.

21.6.3.2.16.5 Xenon

TRACG accounts for the negative reactivity from xenon by adjusting the thermal removal cross section at each node. The PANAC11 wrapup file includes the xenon number density and the microscopic absorption cross section.

In response to RAI 21.6-82, GEH stated that the xenon concentration is not updated during a transient. The staff finds this acceptable for simulating stability, AOO, IE, and ATWS events because the time scales for these events are not long enough for the xenon concentration to change appreciably. However, the startup evolution will take place over the course of hours. In a supplemental response to RAI 21.6-82, the staff requested that GEH justify using the constant xenon assumption for the startup simulation. **RAI 21.6-82 is being tracked as an open item.**

21.6.3.2.16.6 Decay Heat Modeling

LOCA

During an audit of TRACG as applied to ESBWR LOCA, the staff reviewed several documents detailing the procedures and calculations performed to determine the shutdown power curve presented in the ESBWR DCD Tier 2, Revision 2, Figure 6.3-39. The shutdown power is a combination of several heat sources and contributes to the integrated thermal load to be absorbed by the containment. It is also a factor in the determination of margin to specified acceptable fuel design limits during DBAs.

For the ESBWR, the shutdown power is calculated using an offline code and incorporated into TRACG analyses by means of a normalized power table. The TRACG calculation employs the predetermined normalized power table based on a reactor trip signal.

The shutdown power following a SCRAM signal during a design-basis LOCA includes many heat sources, such as those listed below:

- transient fission power during the signal processing and logic delay
- transient fission power during hydraulic control unit valve deenergization and stroke
- transient fission power during control blade insertion
- power from delayed neutron-induced fission
- decay of radioactive fission products
- decay of activated fission products
- decay of actinides in the fuel
- stored energy in the fuel, cladding, vessel, and vessel internals
- decay of activated nuclides in the cladding and other structural materials
- exothermic energy release from water-zirconium reactions

The staff reviewed the specific means employed by GEH for calculating each of these contributions to the total shutdown power.

The staff identified an open item with regard to this particular model for the delayed neutron fission contribution. In particular, the staff cannot conclude that the model is adequately conservative for small-break LOCAs, where the prompt reduction in power during depressurization would not occur.

The acceptability of the DECAY01 generated shutdown power curve is an open item. The staff, however, found that TRACG can be used to verify the conservatism in the DECAY01 generated curve. The team conducted a review of the DECAY01 calculation and requested that GEH perform a confirmatory analysis.

The staff reviewed the conservatisms applied in the analysis, but without specific sensitivities for the type of accident, it is not clear whether the assumption of a large-break LOCA in the fission heat contribution will remain conservative despite the applied increases in irradiation time, exposure, and decreased enrichment. Additionally, while the normalization technique reduces fission power in favor of longer lived decay heat sources during the normalization of the shutdown power shape function, it is not clear whether maximizing each individual component of the decay heat is necessarily conservative during the normalization. For example, artificially increasing the contribution from activation products relative to the delayed fission during the normalization does not appear to necessarily be conservative for all cases.

The staff requested that GEH provide an analysis explicitly calculating the transient fission power to demonstrate that the effects of rapid void formation assumed in DECAY01 do not produce nonconservative estimates of the shutdown power for small-break LOCAs, where rapid void formation does not occur. The staff requested in RAI 6.2-62, that GEH address these issues. **RAI 6.3-62 is being tracked as an open item.**

In Revision 3 of the DCD, GEH states that it is using the 1994 American Nuclear Society's (ANS) decay heat standard. The 1994 decay heat standard is different from that used in Revision 2 of the DCD which states that GEH used the 1979 ANS decay heat standard. DCD Tier 2, Figure 6.3-39 has been updated to show the new decay heat curve. The figures in the DCD demonstrating ECCS performance are unchanged. In RAI 6.3-80, the staff requested additional information from GEH to explain why the analysis plots did not change. In addition, the staff performed a detailed review of the GEH decay heat model during an audit of TRACG as applied to an ESBWR LOCA. RAI 6.3-80 also requests that GEH provide any differences in the methodology that the staff reviewed during the audit and which is being used to generate the decay heat curve in Revision 3 of the ESBWR DCD. **RAI 6.3-80 is being tracked as an open item.**

AOO/IE

For AOO/IE events, TRACG directly calculates the decay heat. The TRACG decay heat model calculates the delayed component of the volumetric heat generation rate in the fuel. The initial nodal decay heat is assumed to be proportional to the fission density in each node. In the transient analysis, TRACG calculates the total nodal power according to the transient flux solution for the fraction of the power produced from fission. This contribution is combined with the decay power predicted using the ANS standard. The total is the transient nodal power. Section 9.3.1 of NEDE-32176P, Revision 3, "TRACG Model Description," April 2006 (NEDE-32176P, Rev. 3), describes this process. The values for the decay heat fractions and the time constants used to calculate the decay heat component are determined from the default May-Witt decay power curves and the 1979 or 1994 ANS standard decay heat models. The user may also specify decay heat group constants through input. For the ESBWR, the 1979 ANS standard is used for transient calculations.

The ANS standards model the total decay heat as the sum of the contributions from fission products, major actinides, miscellaneous actinides, structural activation products, and fission power. The decay heat model does not explicitly account for stored energy since this is accounted for by heat structures in TRACG.

The staff finds that GEH has accounted for all major contributors to the decay heat model adequately. Although the staff did not review the decay heat model in detail, the model is adequate for simulation of AOO and IE events in the ESBWR. All transients that are to be analyzed with the application methodology in Section 4 of NEDE-32176P, Revision 3, experience the limiting conditions before the scram such that the decay heat curve is not as important. The addition of power from decaying isotopes during normal operations is typically on the order of 5–6 percent, and an uncertainty in this small contribution provides little or no effect on the overall transient response. The staff's acceptance of this model for simulating ESBWR AOO/IEs does not constitute acceptance of this model for LOCA applications.

The staff needs to verify that the decay heat model compares with plant data in Revision 3 of NEDE-32177P, "TRACG Qualification." This report was requested in RAI 21.6-75**.** In response, GEH submitted Revision 3 of the TRACG qualification report on August 29, 2007. **RAI 21.6-75 is being tracked as an open item pending staff review of NEDE-32177P, Revision 3.**

Safety Evaluation Report with Open Items for Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design (ML073100168) provide more details on the staff's review of the TRACG decay heat models.

21.6.3.2.17 Numerics

TRACG numerics are a significant improvement over its predecessor, TRAC-BD1/MOD1. As a default, TRACG employs fully implicit integration for hydraulic equations. The fully implicit integration is accomplished by means of a predictor-corrector iterative technique.

TRACG solves the heat conduction equations by implicit integration. The heat transfer coupling between the heat conduction and coolant hydraulics is also treated implicitly via an iterative technique. This implicit coupling represents a significant improvement over commonly used explicit coupling, which may incur an error on the phase shift and amplitude in a thermally induced oscillation.

For the channel components in the time-domain stability analyses, TRACG uses an optional explicit integration because the implicit integration may suppress real physical oscillations. Artificial numerical damping may arise from finite spatial and temporal differencing. This damping is minimized by using an explicit first-order finite differencing method and maintaining the Courant number near 1. The staff found the explicit integration scheme in TRACG an acceptable part of the methodology used for calculating stability margins in ESBWR.

Safety Evaluation Report Regarding the Application of General Electric's Topical Report, "TRACG Application for ESBWR Stability Analysis," NEDE-33083P, Supplement 1 includes more details on the staff's review of the TRACG numerics for stability.

21.6.3.2.18 Component Models

TRACG employs basic component models as building blocks to construct physical models for intended applications. Such an approach renders it a very general and flexible tool to simulate a wide variety of systems. The components that are modeled include pipe, pump, valve, tee, fuel channel, jet pump, steam separator, steam dryer, vessel, upper plenum, heat exchanger, and break and fill as boundary conditions. However, a turbine model for balance of plant (BOP) simulation is missing. The heat exchanger model contains some simplifying approximations that may not be appropriate for simulating the IC or the condenser in BOP. However, GEH is not using the heat exchanger model for simulating either the PCCS or the ICS. TRACG has very sophisticated upper plenum, steam separator, and steam dryer models.

21.6.3.2.18.1 Reactor Vessel

TRACG models the reactor vessel using a three-dimensional VESSEL component. There is one VESSEL component in the LOCA model in which the lower numbered levels of the VESSEL represent the reactor vessel and the higher numbered levels of the vessel represent the containment components. For modeling the reactor vessel, GEH includes channel components, channel bypass, the chimney, and the steam separators.

21.6.3.2.18.1.1 Channel Components

Because of code limitations on the maximum number of components, GEH could not model each channel individually in TRACG. Therefore, GEH combined the channels into groups. The following sections discuss channel grouping and nodalization for the specific ESBWR design-basis analyses.

LOCA

For LOCA applications, the TRACG input deck has three separate rings representing three channel groups within the VESSEL component. The staff finds this representation coarse for traditional LOCA analyses that require modeling of a hot channel. However, since none of the design-basis ESBWR LOCA events show core uncovery, the core does not heat up and GEH did not calculate a PCT, thus making a more detailed channel representation unnecessary. The staff finds the GEH channel grouping acceptable for ESBWR LOCA applications. Stability, AOO/IE, and ATWS events require a more detailed channel representation.

Stability

For stability applications, a fine axial nodalization is used in the core entrance to attempt to maintain a constant Courant number and to provide more detailed modeling of the lower void regions of the core in which void oscillations have the greatest impact on stability. The staff finds the approach described in NEDE-32177P, Revision 2, acceptable.

The channel grouping depends on the type of calculation being performed. For corewide stability analysis, GEH examined the channel grouping described in Figure 5.2-4 of NEDC-33083P, Supplement 1. The staff found that the process for determining adequate channel grouping described in Section 8.1.2.2 of NEDC-33083P, Supplement 1, ensures that spatial variations are modeled adequately for stability analyses.

Regional stability analyses require a different channel grouping to capture the character of higher flux harmonics. Figure 8.1-18 in NEDC-33083P, Supplement 1, depicts this channel grouping. The staff reviewed this procedure and determined that GEH addressed these characteristics and it is adequate for modeling regional oscillations.

The axial and radial nodalizations have been used in the past for TRACG calculations of operating reactor stability and have been found to be adequate for these calculations.

Safety Evaluation Report Regarding the Application of General Electric's Topical Report, "TRACG Application for ESBWR Stability Analysis," NEDE-33083P, Supplement 1, discusses more details of the staff's review of the channel grouping and nodalization for ESBWR stability analyses.

AOO/IE and ATWS

For the ESBWR AOO/IE and ATWS events, GEH used the same axial nodalization for the CHAN component as that used for ESBWR stability analysis (NEDC-33083P, Supplement 1). As discussed in Safety Evaluation Report Regarding the Application of General Electric's Topical Report, "TRACG Application for ESBWR Stability Analysis," NEDE-33083P, Supplement 1, the staff finds this acceptable for use in ESBWR stability analysis. This is also applicable to and therefore acceptable for ESBWR AOO, IE, and ATWS analysis.

GEH states that the channel grouping used for the ESBWR AOO/IE analysis is the same as that described in the TRACG for ESBWR ATWS topical report (NEDE-33083P, Supplement 2). GEH combines the channels based on similar hydrodynamic, as well as neutron kinetics, characteristics. The staff reviewed the GEH channel grouping to verify that it adequately represents the core design as described in NEDC-33239P, Revision 2, "GE14 for ESBWR Nuclear Design Report," April 2007. GEH represented the channels with the highest radial peaking factors as a single channel. In response to RAI 21.6-65, GEH provided a sensitivity study of the channel grouping for the load rejection with total bypass failure infrequent event. GEH demonstrated that a case with more radial channels represented versus the radial grouping used for the licensing-basis ESBWR calculation produces virtually the same results, with the licensing-basis case producing ΔCPR/ICPR results that are more conservative. The staff finds that the GEH channel grouping for performing ESBWR AOO/IE and ATWS calculations as described in NEDE-33083P, Supplement 2, adequately represents the ESBWR core and is acceptable.

The staff requested supplemental information to RAI 21.6-65 on the channel grouping used for ESBWR AOO/IE evaluations. For calculating the ΔCPR/ICPR, the staff asked GEH which channel groups were used. GEH may choose to use only the hot channels. Although this is most conservative in many cases, the staff believes that, for cold water injection events, the ΔCPR/ICPR may be underestimated because the largest change in CPR would take place in the periphery channels. **RAI 21.6-65 is being tracked as an open item.**

21.6.3.2.18.1.2 Chimney

LOCA

During the review of TRACG for ESBWR LOCA, GEH and the NRC staff investigated nodalization and bundle power distributions on the calculated minimum water level in the chimney during a LOCA in the ESBWR. The staff found that calculating minimum water level for the ESBWR LOCA would be most appropriately represented by a single chimney partition. The staff reviewed the TRACG input decks for simulating LOCAs in the ESBWR for design certification and confirmed that GEH had added two individual chimney partitions. GEH used these components to calculate the minimum static head in the chimney.

Stability

In response to staff RAIs, GEH performed a series of detailed analyses of the effect of the chimney on two instability modes—density-wave and loop instability. (See response to RAIs 4.4-11 and 4.4-12 in MFN 06-339). GEH modified the ESBWR TRACG model to include a fine node structure in the chimney region. For the cell size selected by GEH, the time step is limited by the vapor velocity in the chimney (i.e., the Courant number is approximately 1).

The corewide power response to a pressure perturbation was evaluated with the new nodalization. The results were compared to the original (coarse chimney) results in Figure 4.4-11-2 in MFN 06-339. The traces are virtually indistinguishable, thus indicating that fine nodalization in the chimney has no effect on the core decay ratio. Figure 4.4-11-3 in MFN 06-339 compares the core power response to a flow perturbation. Again, there is close agreement between the responses for the coarse and fine nodalizations in the chimney.

Figure 4.4-11-5 in MFN 06-339 shows the results of a channel stability calculation for a high power channel. The calculated results for the case with the finely nodalized chimney compare closely with the original calculation in NEDC-33083P, Supplement 1.

The staff concurs with the GEH evaluation that the finely nodalized chimney allows for a more accurate representation of void propagation through the chimney, but has no effect on the stability results.

Even though the GEH response to the RAI in MFN 06-339 states that, "The original nodalization used for the stability calculations in NEDC-33083P, Supplement 1 and the DCD is adequate for stability analysis," the staff believes that the TRACG model with the fine chimney nodalization should be used for future ESBWR stability calculations. The staff has communicated this to GEH in RAI 4.4-58**. RAI 4.4-58 is being tracked as an open item.**

21.6.3.2.18.1.3 Steam Separator

Since the GEH PIRT does not consider the steam separator to be important, it is modeled by a simple semiempirical model. The model is based on the assumption that the vapor core has solid body rotation and the thin film has azimuthal velocity decaying as the inverse of the square root of the radial position. The model has four constants that are determined by comparing the prediction with the full-scale performance data.

21.6.3.2.18.2 Isolation Condenser System

During a LOCA, the ICS provides additional liquid inventory to the RPV upon opening of the condensate return valves to initiate the system. For AOO/IE and ATWS events, the ICS also provides the reactor with depressurization for pressurization events such as an MSIV closure. GEH also simulated ICS heat removal during a LOCA event.

The IC testing was performed at the PANTHERS-IC test facility. Section 4.2 of NEDC-32725P, Revision 1, describes the test, along with the TRACG comparisons. Section 3.2 of MFN 04-059 includes the updated comparisons with TRACG04. The staff reviewed this information in conjunction with the additional information GEH provided in response to RAI 21.6-55.

GEH performed a series of steady-state and transient tests and compared these data using TRACG simulations. The steady-state tests were performed to test the intended operation of the IC, such as condensing steam during a reactor isolation event. The transient tests simulated abnormal IC operations, including noncondensible gas buildup and a pool water level transient. The staff reviewed the results of these tests and the modeling of the ICS in the ESBWR TRACG input decks. Safety Evaluation with Open Items, Application of the TRACG Computer Code to Anticipated Operational Occurances and Infrequent Events for the ESBWR Design NEDE-33083P, Chapter 4," (ML073100168) documents in detail the staff's review.

GEH indicated in respone to RAI 21.6-55 and in Section 4.4.1 of NEDC-33083P-A that the AOO/IE and ATWS model uses a different heat transfer correlation than that used to model the PANTHERS and PANDA facilities for the secondary-side heat transfer. The staff requested that GEH explain and justify the differences between the modeling strategy for the pool side and that of PANTHERS and the AOO/IE and ATWS model. The staff requested this information as a supplement to RAI 21.6-55. **RAI 21.6-55 is being tracked as an open item.**

Although the staff finds that TRACG is capable of modeling the IC behavior during AOO/ATWS and LOCA events for purely steam condensation, the staff cannot determine whether this modeling is accurate or conservative during the presence of noncondensible gases. The staff agrees with GEH that the time scales seen during an AOO/ATWS event are short enough that noncondensible gases generated by radiolytic decomposition will be insignificant and that modeling of pure steam condensation for these events is appropriate. However, the staff does find that the time scales of interest for modeling a LOCA event warrant the inclusion of the noncondensible radiolytic gases in the model, and GEH does include noncondensible gases into its IC model for this purpose. However, the staff does not have enough information to determine whether this modeling is accurate or conservative. In addition, since the modeling of the ICS used for ESBWR AOO/ATWS and LOCA analyses is different than that used in the PANTHERS simulations, the staff cannot determine whether this modeling will adequately represent the ESBWR IC. In the RAI supplement to RAI 21.6-55 the staff asked GEH to address these concerns.

21.6.3.2.18.3 Standby Liquid Control System Modeling

The staff reviewed the GEH modeling of the SLCS for LOCA and ATWS applications. In RAI 21.6-12, the staff asked GEH to justify the velocity it selected for this component. The staff found potential errors in the response, and requested supplemental information. **RAI 21.6-12 is being tracked as an open item.**

21.6.3.2.18.4 Containment Components

The staff's evaluation of TRACG for containment analysis is documented in the SER approving NEDC-33083P-A. However, the TRACG model nodalization has been modified to incorporate the design changes, additional features, and finer details compared to the one used in its preapplication review. For ESBWR LOCA applications, GEH combined the detailed reactor pressure vessel (RPV) and the detailed containment model into a single consistent input deck. The same deck is exercised for both containment and water level calculations. The air gap between the reactor shield wall and the pressure vessel is also modeled in the combined TRACG nodalization. The RPV wall is modeled as a lumped heat slab. The combined TRACG model results compared well with those from the base cases, and the impacts due to nodalization changes on the minimum chimney static head level and the long-term drywell pressure are judged to be small compared to the margins. The TRACG model is currently setup for 4500 MW core power. These design and modeling changes are documented in Table 6A-1, Table 6.2-6a, and Appendix 6B of DCD Tier 2, Revision 4. The staff's evaluation of these changes is included in Section 6.2.1 of this report.

Section 4.0 of the staff SER documented in NEDC-33083P-A includes 20 "Confirmatory Items" that were identified as needing resolution at the design certification stage. The staff's evaluation of these confirmatory items is included in the Addendum to the Safety Evaluation Report with Open Items for NEDC-33083P-A, Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design (ML073190044). As discussed in the addendum, in regard to staff's review of TRACG as applied to containment, RAI 6.2-98, RAI 21.6-69, RAI 21.6-71, RAI 21.6-75 are currently being tracked as open items. RAI 6.2-98 S01 is tracking questions regarding the MSLB and FWLB nodalization and results differences, usage of a double pipe to purge the non-condensable gases from the GDCS airspace, and the newly proposed drywell gas recirculation system. RAI 21.6-69 S01 is tracking questions regarding the sensitivity of TRACG results to PIRT multipliers and radiolysis. RAI 21.6-71 S01 is tracking queries to assess TRACG's ability to simulate the purging of non-condensable gases from the DW and GDCS airspace to WW airspace, at 72 hours. RAI 21.6-75 tracks the staff's review of Revision 3 of the TRACG Qualification report.

Hereunder are described the various TRACG containment components and their TRACG treatment. The staff is evaluating the adequacy of the containment models for calculating containment peak pressure as discussed in Chapter 6.2 of this report.

21.6.3.2.18.4.1 Wetwell

The wetwell consists of the suppression pool (SP) and wetwell gas space. The wetwell is bounded by the diaphragm floor on top, containment outer wall, and wetwell inner wall on the sides and the floor of the containment. During blowdown, flow from the safety/relief valves (SRVs) is directed to the SP and quenched via the SRV discharge lines. Flow from the LOCA break and DPVs is directed from the drywell to the SP and quenched via the SP horizontal vent system. Any flow through the passive containment cooling (PCC) vents is also discharged to the SP.

Wetwell Gas Space

The wetwell gas (steam and non-condensables) space is also represented by multidimensional cells. Typically, rings and axial levels are employed in the TRACG model, which allows for natural circulation in this region. The flow regimes in this region will be the same as in the drywell— single-phase gas, dispersed droplets resulting from entrainment from the SP, and a condensate film on the walls. The models involved in the calculation include turbulent shear between cells, non-condensable distribution, wall friction, interfacial friction, wall heat transfer, fogging and interfacial heat transfer, and heat transfer at the SP interface.

Suppression Pool

The SP is also represented by multidimensional cells. Rings and axial levels are used to represent the pool. The major phenomena of interest for the SP include steam condensation with or without non-condensable gases, temperature distribution, thermal stratification, and pool two-phase level.

21.6.3.2.18.4.2 Passive Containment Cooling Pools

The six PCC pools are located outside (above) the containment. The total PCCS cooling capacity is 66 MW. Each contains one PCC unit. The pools are interconnected with each other and with the IC pools.

The pools are represented as part of the three-dimensional TRACG region, partitioned into the IC and PCC pools. The pools are allowed to communicate with each other at the bottom and the top. The pools are modeled with rings and axial levels. Heat transfer occurs from the PCC headers and tubes to the water in the pools. Pool side heat transfer is calculated using correlations either as boiling heat transfer or as single phase convection to liquid for subcooled conditions. The staff reviewed the pool heat transfer correlations used for both boiling heat transfer and single-phase convection to liquid and found them acceptable for this application. The staff's safety evaluation of NEDC-33083P-A, "TRACG Application for ESBWR," documents this review.

21.6.3.2.18.4.3 Passive Containment Cooling Units

The ESBWR has six PCC heat exchanger units. Each is comprised of two-module drum and tube heat exchangers using horizontal upper and lower drums connected with a multiplicity of vertical tubes (280 tubes per module). Two identical modules are coupled to form one PCC heat exchanger unit. The PCC units are represented by one-dimensional components simulating the inlet piping, headers, condenser tubes, condensate discharge lines, and vent lines. One-dimensional forms of the mass, momentum, and energy equations are applicable. Heat is transferred through the walls of the tubes and headers to the respective pools.

The PCCS can operate in two distinct modes—a condensation mode and a pressure differential mode. In the condensation mode, the steam is condensed in the vertical tubes and the condensate is drained from the lower drum to the individual unit's drain tank. In the pressure differential mode, the flow through the PCC heat exchangers is caused by a drywell-wetwell pressure difference. Since the PCC vent line outlet is higher than the outlet of the upper horizontal drywell/wetwell LOCA vents. Non-condensable gases and uncondensed steam are vented to the suppression pool. The staff reviewed the correlation for calculating heat transfer inside the tubes as well as the configuration of the PCCS, and found them acceptable for this application. The staff's safety evaluation of NEDC-33083P-A, "TRACG Application for ESBWR," documents this review.

21.6.3.2.18.4.4 Horizontal Vent System

The ESBWR has 36 horizontal vents between the drywell and the SP. There are 12 vertical flow channels, and each contains three horizontal vents attached to a vertical vent pipe. The top row of horizontal vents is approximately 0.9 meters below the bottom of the PCC vents. The remaining two rows of vents are each vertically separated by 1.37 meters. GEH models the horizontal vents in TRACG. The staff's safety evaluation of NEDC-33083P-A, "TRACG Application for ESBWR," documents the review of the horizontal vents.

21.6.3.2.18.4.5 GDCS Equalizing Lines

Four GDCS equalizing lines (one per division) connect the SP to the RPV downcomer. During the long-term cooling phase of the post-LOCA transient, the squib valves in these lines will open if the RPV level in the downcomer drops to 1 meter above the top of the active fuel or 8.453 meters from the RPV bottom, with a 30-minute delay time to create a permissive signal. As stated by GEH in response to RAI 6.3-40, these valves do not actuate during any design-basis LOCA event for 72 hours. The TRACG model for an ESBWR LOCA contains a model for the equalizing lines. The correlations used for wall friction and singular losses are the same as used for the horizontal vents. The staff's safety evaluation of NEDC-33083P, "TRACG Application for ESBWR", documents the review of the GDCS equalizing lines.

21.6.3.2.18.4.6 Vacuum Breakers

The ESBWR has three VBs connecting the upper drywell to the wetwell gas space. The VBs will open when the pressure in the wetwell is higher than that of the drywell by a specified value. The VBs are represented by one-dimensional VALVE components. VBs are lumped together as one component. The VBs are triggered open at a set negative pressure differential between the drywell and wetwell. They will close at a lower value of the pressure differential. The VBs transport flow from the wetwell gas space to the drywell at conditions corresponding to the cell in the wetwell gas space to which they are connected. The correlations used for the singular losses are the same as for the horizontal vents. The staff's safety evaluation of NEDC-33083P, "TRACG Application for ESBWR", documents the review of the vacuum breakers.

21.6.3.2.18.4.7 RPV Level vs. Minum Containment Pressure

During the December 2006 audit, the NRC staff and GEH discussed the use of TRACG Containment model in the RPV level calculation. The NRC staff was concerned that the containment portion of the input deck was designed with assumptions to give maximum containment pressure and this may be non-conservative for RPV level calculations as historical operating plant analyses have been performed with minimum containment pressure. It was not concluded at that time that what the impact of minimum containment pressure would have on the ESBWR RPV level calculation. This issue was addressed in RAI 6.2-144. In response, GEH evaluated the impact of containment back pressure on the ECCS performance and presented in ESBWR DCD Tier 2, Revision 4, Appendix 6C. The staff reviewed GEH's evaluation and determined that the minimum chimney collapsed level is not sensitive to the changes in the containment back pressure expected for the ESBWR design under LOCA conditions.

21.6.3.2.18.4.8 TRACG modeling of Steam Source

During the December 2006 audit, the GEH showed the results of sensitivity studies where GEH examined the effects of placing the steam source in different elevations of the drywell. GEH showed that the highest peak pressure results from placing the steam source at the highest elevation. The staff noted that it is inconsistent with current GEH practice of placing the steam source below the RPV. GEH addressed this issue in response to RAI 6.2-53, Supplement 1. In this response, GEH showed sensitivity analyses results to confirm that the current bounding MSLB case where break occurs at Level 34 is limiting. For calculations in DCD Revision 2, GEH has used the MSLB break location at Level 23, which the staff showed as not conservative during the December 2006 audit. As a result, GEH has changed the MSLB break location to Level 34 in DCD Revision 3.

21.6.3.3 Accident Scenario Identification Process

The behavior of a nuclear power plant undergoing an accident or transient is not influenced in an equal manner by all phenomena that occur during the event. A determination needs to be made to establish those phenomena that are important for each event and the various phases within an event. Development of a PIRT establishes those phases and phenomena that are significant to the progress of the event being evaluated.

21.6.3.3.1 LOCA

Important phenomena for LOCAs in the ESBWR have been identified in two PIRTs—LOCA/ECCS and LOCA/containment. The PIRT for LOCA/ECCS includes all the high- and medium-ranked phenomena in the RPV, main steamlines, and ICS, including system interactions. The PIRT for LOCA/containment covers all the high- and medium-ranked phenomena in the drywell, wetwell, GDCS, PCCS, DPVs, VBs, main vents (between the drywell and wetwell), and SRV quenchers. A team of experts conducted both top-down and bottom-up processes to obtain these phenomena, which were later used for TRACG assessment.

The two PIRTS were further evaluated and revised according to the results of the scaling analysis. The revision either confirmed or downgraded, with few exceptions, the ranking of some high-ranked phenomena. However, the revision upgraded several events to high-ranked phenomena, including the addition of the mass flow through the break during the GDCS injection and long-term cooling phases of the LOCA, mass flow through the SRVs/DPVs during the GDCS injection phase of the LOCA, flashing/redistribution in the control rod guide tube region, and flashing in the downcomer annulus during the GDCS injection phase of the LOCA.

The staff found the PIRT for ESBWR LOCA acceptable. NEDC-33083P-A includes more details of the staff’s review of the PIRT for LOCA.

21.6.3.3.1.1 LOCA Long-Term Core Cooling

GEH submitted details on long-term core cooling in MFN 05-105. This reference included a discussion on long-term inventory distribution for four break locations—(1) MSLB, (2) FWLB, (3) BDLB, and (4) GDCS line break.

GEH provided a PIRT related to long-term core cooling of the ESBWR. The phenomena that ranked high for the MSLB and FWLB were decay heat, GDCS pool volume versus elevation, and RPV volume versus elevation. GEH gave PCCS capacity a ranking of medium for these events. The phenomena that ranked high for the BDLB and GDCS line break were decay heat, DPVs (break flow and pressure drop), PCCS capacity, lower drywell volume versus elevation, GDCS pool volume versus elevation, and RPV volume versus elevation.

The staff's Addendum with Open Items to the Safety Evaluation NEDC-33083-P-A “TRACG Application for ESBWR” (ML073190044) provides more details of the staff’s review of the long-term core cooling PIRT.

There will be some condensation of the steam in the drywell that may affect the volume of water that will return to the vessel from the PCCS drainline. The TRACG ESBWR containment model was designed to maximize peak pressure and does not contain heat sinks and so may not accurately calculate this condensation. The staff believes that the amount of condensation in the drywell should be small in comparison to the condensation in the PCCS. However, the staff requested in RAI 6.2-144 that GEH investigate the effects of assuming lower pressure in the drywell on RPV level calculations. This study should confirm that such an assumption does not have a large impact on long-term core cooling. **RAI 6.2-144 is being tracked as an open item.**

The staff has concerns about the ability of TRACG to track noncondensible gases, particularly whether TRACG can calculate the right amount of noncondensible gases that enter the PCCS. The staff requested GEH to address these concerns in its response to RAI 21.6-96.
RAI 21.6-96 is being tracked as an open item.

The NRC staff requested that GEH demonstrate that the core remains covered for 72 hours for all breaks in RAI 6.3-64. **RAI 6.3-64 is being tracked as an open item.**

21.6.3.3.2 Stability

A PIRT identified important phenomena for stability in the ESBWR. The PIRT for stability includes all the high- and medium-ranked phenomena. The staff found the stability PIRT to be comprehensive, giving the appropriate ranking to stability phenomena. Safety Evaluation Report Regarding the Application of General Electric's Topical Report, "TRACG Application for ESBWR Stability Analysis," NEDE-33083P, Supplement 1, contains the details of the staff's review of the PIRT for ESBWR stability.

21.6.3.3.3 AOO/IE

GEH identified important phenomena for AOOs in the ESBWR in a PIRT. The transient events have been categorized into three groups—(1) pressurization events, (2) depressurization events, and (3) cold water insertion events. For each event type, the phenomena are listed and ranked for each major component in the reactor system. The staff's reviewed the AOO PIRT is in the Safety Evaluation Report with Open Items for Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design (ML073100168).

In RAI 21.6-61, the staff questioned the ranking of mixing in the lower plenum for cold water injection events. The GEH response included a nodalization study of the feedwater controller failure (FWCF) event to maximum demand. The staff is concerned that, as cold water enters the core, it might not mix well in the lower plenum and more cold water may be concentrated at the periphery causing a more pronounced effect on ΔCPR for bundles in this location. GEH performed a nodalization study of the FWCF event in which it increased and decreased the rate of transfer between the radial rings in the lower plenum by artificially creating resistance between these cells. GEH showed that the ultimate change in ΔCPR, given the different resistances in the lower plenum, did not have a substantial effect on ΔCPR for Ring 1 (the central most) and 2 (next to central most). GEH did not display the results of the ΔCPR changes for Ring 3, which is the outermost ring and the one of most concern. In addition, the staff is aware that GEH is planning to redesign the feedwater heating system so that the LFWH event would create more severe cold water transient. The staff will review Revision 4 of the DCD to determine if the additional information submitted by GEH in Revision 4 will address this concern. In addition, since there are design changes in DCD Revision 4, the staff may need to revisit this issue.

21.6.3.3.4 ATWS

GEH identified important phenomena for ATWS in the ESBWR in a PIRT. The phenomena are identified as having an effect on three critical safety parameters—(1) SP temperature, (2) vessel pressure, and (3) fuel clad temperature.

In ranking the phenomena, GEH divided the limiting scenarios into five phases—(1) short-term pressurization, neutron flux increase, and fuel heatup, (2) feedwater runback and water level reduction, (3) boron injection, mixing, and negative reactivity insertion, (4) postshutdown SP heatup, and (5) depressurization of the reactor.

The ESBWR ATWS PIRT builds upon the ESBWR AOO PIRT. Several PIRT parameters were introduced specifically for the ESBWR ATWS evaluation including the following:

- ATW1: boron mixing/entrainment between the jets downstream of the injection nozzle
- ATW2: boron settling in the guide tubes or lower plenum
- ATW3: boron transport and distribution through the vessel, particularly in the core bypass region
- ATW5: boron reactivity

The staff finds that the ATWS PIRT is comprehensive and gives the appropriate rating to the phenomena important for ESBWR ATWS. Safety Evaluation with Open Items. Application of the TRACG Computer Code to Anticipated Transients without Scram for the ESBWR Design NEDE-33083P, Supplement 2," (ML073190036) discusses the staff's review.

21.6.3.4 <u>Code Assessment</u>

The staff reviewed the following assessment cases that were performed using TRACG04:

- separate effects, component, and integral tests
    - Toshiba low-pressure tests
    - Ontario Hydro large diameter tests
    - PANTHERS/PCCS tests
    - PANTHERS/ICS tests
    - SP
    - FIST LPCI break results
    - GIRAFFE systems interaction (GS1) test
    - GIRAFFE/Helium test
    - ROSA-IV LOCA tests
    - One-sixth-scale boron mixing
    - PSTF containment response tests
    - FIX-II LOCA tests
    - PANDA transient tests

- Plant Data

  - Dodewaard startup (natural circulation)
  - Peach Bottom stability tests
  - NMP-2 instability event

The staff audited these assessments. Other tests have been reviewed as part of the TRACG review and are documented in other sections of this report as well as in NEDC-33083P-A, Safety Evaluation Report Regarding the Application of General Electric's Topical Report, "TRACG Application for ESBWR Stability Analysis," NEDE-33083P, Supplement 1 and NEDE-32906P, Supplement 3. The staff confirms that GEH has extensively qualified TRACG04 and that TRACG04 adequately predicts all highly ranked behavior. GEH submitted Revision 3 of the ESBWR Qualification report in response to RAI 21.6-75 that employed the TRACG04 version used to perform all ESBWR LOCA events in the design certification documentation. This document is currently under review. **RAI 21.6-75 is being tracked as an open item.**

21.6.3.5 Uncertainty Analysis

21.6.3.5.1 LOCA

In MFN 05-096, GEH states that, since there is no core heatup, an uncertainty analysis on PCT would not provide useful results. GEH further states that a bounding evaluation for the minimum water level in the chimney during a LOCA event would demonstrate that there is margin to core uncovery and heatup. As stated in 10 CFR 50.46(a)(1)(i), "comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for…." Furthermore, 10 CFR 50.46(a)(1)(ii) states, "Alternately, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models." The staff issued RAI 6.3-81 requesting that GEH demonstrate how the LOCA analyses comply with this requirement. **RAI 6.3-81 is being tracked as an open item.**

21.6.3.5.2 Stability

GEH accounts for the uncertainty in the stability calculations in two ways. First, for the ESBWR it sets the design criteria for the decay ratio to 0.8 to allow 0.2 in uncertainty. In addition, GEH statistically combines the uncertainty of all of the medium and highly ranked phenomena. GEH chose the normal distribution-one-sided upper tolerance limit (ND-OSUTL) method if the output distribution is normal, otherwise GEH used the order statistics method. The staff reviewed and accepted the 1σ uncertainty values of the high- and medium-ranked parameters, which are discussed in Safety Evaluation Report Regarding the Application of General Electric's Topical Report, "TRACG Application for ESBWR Stability Analysis," NEDE-33083P, Supplement 1. The staff reviewed the uncertainties in the physics parameters as part of design certification; Addendum with Open Items to the Safety Evaluation, Application of the TRACG Computer Code to Stability Analysis for the ESBWR Design NEDE-33083P, Supplement 1, (ML073190079) discusses these uncertainties.

21.6.3.5.3 AOO/IE

For the TRACG application for ESBWR AOO analysis described in NEDC-33083P-A, GEH chose the ND-OSUTL method if the output distribution is normal, otherwise GEH used the order statistics method.

This method for determining combined bias and uncertainty is the same as that used in the GEH application of TRACG to the BWR/2-6 AOO analysis (NEDE-32906P-A, Rev. 2,) and to ESBWR stability (NEDC-33083P, Supplement 1). This method was reviewed in detail during the review of the applicability of TRACG to BWR/2-6 AOO analysis. The associated SER documents the staff's review. The staff finds the use of this methodology acceptable for determining the combined uncertainty for TRACG modeling of the ESBWR AOO events.

GEH accounts for the uncertainty of the highly ranked parameters. The staff's review of 1σ uncertainty values of the highly ranked parameters is discussed in staff's Safety Evaluation Report with Open Items for Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design (ML073100168).

The difference between the GEH application of this methodology and that of NEDC-33083P, Supplement 1, is that GEH only includes the highly ranked PIRT phenomena in the application of TRACG to ESBWR AOOs, whereas GEH includes high- and medium-ranked phenomena for the other approved applications. The staff asked GEH in RAI 21.6-64 to justify the exclusion of the medium-ranked phenomena. **RAI 21.6-64 is being tracked as an open item.**

21.6.3.5.4 ATWS

GEH uses a different method for combining uncertainties for the ATWS events. The staff reviewed the method and found it acceptable for ESBWR ATWS events. Safety Evaluation with Open Items, Application of the TRACG Computer Code to Anticipated Transients Without Scram for the ESBWR Design NEDE-33083P, Supplement 2, (ML073190036) discusses this finding in greater detail.

GEH accounts for the uncertainty of the highly ranked parameters. The staff reviewed the 1σ uncertainty values of the highly ranked parameters, which are discussed in Safety Evaluation with Open Items, Application of the TRACG Computer Code to Anticipated Operational Occurances Without Scram for the ESBWR Design NEDE-33083P, Supplement 2, (ML073190036).

21.6.4 Staff Calculations

The staff made independent calculations of ESBWR events using the TRACE thermal-hydraulic code coupled to the PARCS neutronic code. The models were developed using ESBWR design information.

The staff has performed preliminary calculations for the LOCA water level events. However, TRACE input decks are still under development during the drafting of this SER. **This is an open item.**

21.6.5 Conclusions

- Due to the open items that remain to be resolved for this section the staff was unable to finalize its conclusions regarding acceptability

21.7 Quality Assurance Inspection

The staff relied upon four principle test programs (PANDA, PANTHERS/PCCS, PANTHERS /ICS and GIRAFFE) to demonstrate that ESBWR accident analyses analytical methods and computer codes described in Section 21 of this report were adequately and appropriately applied. The purpose of the test programs was to validate the capability of accident analysis computer models to predict plant thermal hydraulic behavior for a variety of accident conditions. These test programs were originally conducted to support the design certification application for the SBWR design.

During the 1994 -1996 time frame, the staff conducted QA program implementation inspections of GE's major SBWR design certification thermal-hydraulic test programs used in the design and licensing of the SBWR. The NRC staff inspections were conducted to determine if GEH and it's contractors/partners fulfilled GEH's commitment to their DCD Chapter 17 QA requirements. These test programs are described in detail in the ESBWR DCD Tier 2 Section 1.5, "Requirements for Further Techncial information," and also in Section 21.3, "Overview of General Electric Testing Programs," of this report. The NRC inspections of the GEH test programs resulted in the identification of findings (Notice of Nonconformance and Unresolved Items) for failure to adequately implement QA program requirements, procedural requirements and testing activities in certain cases. GEH provided responses to these NRC inspections findings that described the GEH corrective actions. These responses and corrective actions were accepted by the staff as responsive to the NRC inspection findings.

As described in Section 17.0 of this report, the applicant has continuously maintained a QA program meeting the requirements of Appendix B to 10 CFR Part 50 that spanned SBWR and ESBWR design activities. Because testing activities used to confirm the validity of safety-related analytical methods and computer codes are within the scope of Appendix B QA requirements, the staff reviewed the QA program controls applied to testing activities. Specifically, the staff verified that test facilities implemented QA controls which met the requirements of the applicant's QA program and Appendix B to 10 CFR Part 50 during the SBWR design certification review. As part of the QA review, the staff reviewed the QA program used at each of the test facilities and performed inspections to verify that the quality program was effectively implemented. The last of these inspections was performed at PANDA in 1996. For each of the four SBWR test programs, the staff concluded that reasonable and appropriate QA measures were used to control SBWR test activities. See resolution of URI 05200010-2005-201-02 below.

The staff performed inspections of the implementation of the GEH QA program on the ESBWR activities as part of the staff's review of DCD Chapter 17. These staff inspections were performed in November 2005, April 2006, and December 2006.

- David B. Matthews, Nuclear Regulatory Commission, to David H. Hinds, GE Nuclear Energy, "NRC Inspection Report 05200010/2005-201 and Notice of Nonconformance" January 11, 2006 (ADAMS Accession No. ML053560155).

- David B. Matthews, Nuclear Regulatory Commission, to David H. Hinds, GE Nuclear Energy, “NRC Inspection Report 05200010/2006-201 and Notice of Nonconformance.” June 14, 2006. (ADAMS Accession No. ML061590328).

- David B. Matthews, Nuclear Regulatory Commission, to David H. Hinds, GE Nuclear Energy, “NRC Inspection Report for General Electric Nuclear Energy (GENE) General Economic and Simplified Boiling Water Reactor (ESBWR) Quality Assurance Implementation Follow-up Inspection.” January 19, 2007. (ADAMS Accession No. ML070100142).

As part of the November 2005 inspection, the staff identified URI 05200010-2005-201-02 concerning the previous SBWR Qualification Test Program Quality Assurance Inspections. This URI requested that GEH recapture all the inspection documentation records related to the GEH SBWR design certification testing programs performed that are being used to support design certification of the ESBWR.

GEH responded in the following letters:

- David H. Hinds, GE Nuclear Energy, to Document Control Desk, U.S. Nuclear Regulatory Commission, “Reply to Notice of Nonconformance NRC Inspection Report 05200010/2005-201, dated January 11, 2006.” February 9, 2006. (ADAMS Accession No. ML060440566).

The October 27, 2006, response described the quality oversight activities that had been performed and identified the NRC and GEH documentation that supported the SBWR test program inspections. Enclosure 2, “Quality Oversight for the SBWR Test Program,” to MFN 06-053 Supplement 1, included the description and identification of the oversight documentation for the PANDA, PANTHERS/PCCS, PANTHERS/ICS and GIRAFFE test programs. Based on review of the documentation supplied by GEH, the staff closed URI 05200010-2005-201-02:

- David H. Hinds, GE Nuclear Energy, to Document Control Desk, U.S. Nuclear Regulatory Commission, “NRC Inspection 05200010/2005-201 Unresolved Items.” October 27, 2006. (ADAMS Accession No. ML063170069).

Conclusions

Because the accident analysis computer models have not changed since they were originally validated during the SBWR design certification review, the staff concludes that the previous QA test program reviews conducted to support SBWR design certification generally remain valid for the ESBWR design. After considering the QA implementation inspections of GEH's design certification test facilities and programs, the staff concluded that the QA programs governing GEH's SBWR/ESBWR design certification test programs satisfied the requirements of 10 CFR Part 52 and the pertinent provisions of Appendix B to 10 CFR Part 50.

21.8 Reference

GE Nuclear Energy, "ESBWR Test and Analysis Program Description," NEDC-33079P, Class III, Revision 1, November 2005.

Letter from W. D. Beckner (NRC) to L. M. Quintana (GENE), "Re-Issuance of Safety Evaluation Report Regarding the Application of General Electric Nuclear Energy's's TRACG Code to ESBWR Loss-of-Coolant (LOCA) Analyses," October 24, 2004.

GE Nuclear Energy, "TRACG Qualifications," NEDE-32177P, Class III, Revision 2, January 2000.

GE Nuclear Energy, MIT and UCB Separate Effects Tests for PCCS Tube Geometry, "Single Tube Condensation Test Program," NEDC-32301, March 1994.

GE Nuclear Energy, "Simplified BWR Program Gravity-Driven Cooling System (GDCS) Integrated Systems Test," GEFR-00850, October 1989.

GE Nuclear Energy, "SBWR Testing Summary Report," NEDC-32606P, Class III, November 1996.

GE Nuclear Energy, "ESBWR Test Report," NEDC-33081P, Class III, Revision 1, May 2005.

GE Nuclear Energy, "Scaling of the SBWR Related Tests," NEDC-32288P, Class III, Revision 1, October 1995.

GE Nuclear Energy, "ESBWR Scaling Report," NEDC-33082P, Revision 1, Class III, January 2006.

MFN 06-225, "Response to NRC Request for Additional Information Letter No. 4 Related to ESBWR Design Certification Application – ESBWR Scaling Analysis – RAI Number 6.3-1," July 18, 2006.

NUREG/CR-5809, *A Hierarchical, Two-Tiered Scaling Analysis*, November 1991.

M. di Marzo, "A Simplified Model of the BWR Depressurization Transient," Nuclear Engineering and Design, 205 (2001), pgs. 107-114, July 28, 2000.

MFN 05-109, "GE Response to Results of NRC Acceptance Review for ESBWR Design Certification Application - Item 2," October 20, 2005.

MNF 276-95, "GIRAFFE/SIT Trip Report dated 10/27/95," November 7, 1995.

MNF 170-94, "Summary of the visit on October 16, 1994, at the Societa Informationi Esperienze Termoidrauliche (SIET) Performance Analysis and Testing of Heat Removal System (PANTHERS) Test Facility for the SBWR Design," December 21, 1994.

NEDC-33083P-A, MFN 05-017, "TRACG Application for ESBWR," March 2005, (ADAMS Accession No. ML051390265).

NEDC-33083P, Supplement 1, “TRACG Application for ESBWR Stability,” December 9, 2004, (ADAMS Accession No. ML0500301602).

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. ML072270267 and ML072270276).

NEDE-33083P, Supplement 2, “TRACG Application for ESBWR Anticipated Transient Without Scram Analyses,” January 2006, (ADAMS Accession No. ML060190592).

MFN 05-096, Hinds, D.H. General Electric, Letter to U.S. Nuclear Regulatory Commission, “Summary of September 9, 2005 NRC/GE Conference Call on TRACG LOCA SER Confirmatory Items,” September 20, 2005, (ADAMS Accession No. ML052910378).

MFN 05-105, Hinds, D.H., General Electric, Letter to U.S. Nuclear Regulatory Commission, “TRACG LOCA SER Confirmatory Items (TAC # MC868), Enclosure 2, Reactor Pressure Vessel (RPV) Level Response for the Long Term PCCS Period, Phenomena Identification and Ranking Table, and Major Design Changes from Pre-Application Review Design to DCD Design,” October 6, 2005, (ADAMS Accession No. ML053140223).

NUREG/CR-5249, “Quantifying Reactor Safety Margins: Application of Code Scaling Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident,” December 1989, (ADAMS Accession No. ML070310119).

MFN 05-084, Hucik, S.A., General Electric, Letter to W.D. Beckner, U.S. Nuclear Regulatory Commission, “General Electric Company Application for Final Design Approval and Design Certification of ESBWR Standard Plant Design,” August 24, 2005, (ADAMS Accession No. ML052490334).

NEDE-33038P, MFN 04-109, Section 4.7, “Demonstration Calculations for ESBWR AOOs,” October 8, 2004, (ADAMS Accession No. ML042930020).

NEDE-32176P, Revision 3, “TRACG Model Description,” April 2006, (ADAMS Accession No. ML061160238).

NEDE-32906P-A, Revision 2, MFN 06-046, “TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis,” February 28, 2006, (ADAMS Accession Nos. ML060530571 and ML060530575).

Draft Regulatory Guide, DG-1120, “Transient and Accident Analysis Methods,” U.S. Nuclear Regulatory Commission, December 2000, (ADAMS Accession No. ML052560476).

Draft Standard Review Plan, Section 15.0.2, “Review of Analytical Computer Codes,” U.S. Nuclear Regulatory Commission, December 2000, (ADAMS Accession No. ML003773356).

NUREG-0737, “Clarification of TMI Action Plan Requirements,” November 1980, (ADAMS Accession No. ML051400209).

NEDC-33079P, Revision 1, “ESBWR Test and Analysis Program Description,” March 2005, (ADAMS Accession No. ML051390233).

NEDE-32177P, Revision 2, “TRACG Qualification,” January 2000, (ADAMS Accession No. ML003683162).

MFN 04-059, “Update of ESBWR TRACG Qualification for NEDC-32725P and NEDC-33080P Using the 9-Apr-2004 Program Library Version of TRACG04,” June 6, 2004, (ADAMS Accession No. ML041610037).

NEDC-32725P, Revision 1, “TRACG Qualification for SBWR,” August 30, 2002, (ADAMS Accession Nos. ML022560558 and ML022560559).

NEDE-32176P, Revision 2, “TRACG Model Description,” December 1999, (ADAMS Accession No. 993630283).

NEDE-32906P, Supplement 3, “Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,” May 2006, (ADAMS Accession No. ML070300348).

NEDC-33239P, Revision 2, “GE14 for ESBWR Nuclear Design Report,” April 2007, (ADAMS Accession No. ML072430810).

UM-0136, Revision 0, “TRACG04A, P User’s Manual,” December 2005, (ADAMS Accession No. ML071150376).

Addendum to the Safety Evaluation Report with Open Items for NEDC-33083P-A, Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design (ML073190044)

Addendum to the Safety Evaluation Report (NEDC-33083P, Supplement 1) for TRACG as applied to Stability (ML073190079)

Safety Evaluation Report with Open Items for Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design (ML073100168)

Safety Evaluation with Open Items for Application of the TRACG Computer Code to Anticipated Transients Without Scram for the ESBWR Design NEDE-33083P, Supplement 2," October 2007 (ML073190036)

MFN 05-133, Stramback, G., General Electric, Letter to U.S. Nuclear Regulatory Commission, “Response to DSS-CD TRACG LTR RAIs,” November 11, 2005.

“Safety Evaluation Report by the Office Of Nuclear Reactor Regulation Application for GE14 for ESBWR Nuclear Design Report (NEDC-33239P) and Gamma Thermometer System for LPRM Calibration and Power Shape Monitoring (NEDE-33197P) LTRs for Reference in the ESBWR Design Certification Application (Tac No. MD1464).” ***[report is still in draft form, and reference will be provided in final SER]***

David B. Matthews, Nuclear Regulatory Commission, to David H. Hinds, GE Nuclear Energy, "NRC Inspection Report 05200010/2005-201 and Notice of Nonconformance," January 11, 2006. (ADAMS Accession No. ML053560155).

David H. Hinds, GE Nuclear Energy, to Document Control Desk, U.S. Nuclear Regulatory Commission, "Reply to Notice of Nonconformance NRC Inspection Report 05200010/2005-201, dated January 11, 2006," February 9, 2006. (ADAMS Accession No. ML060440566).

David B. Matthews, Nuclear Regulatory Commission, to David H. Hinds, GE Nuclear Energy, "NRC Inspection Report 05200010/2006-201 and Notice of Nonconformance," June 14, 2006. (ADAMS Accession No. ML061590328).

David H. Hinds, GE Nuclear Energy, to Document Control Desk, U.S. Nuclear Regulatory Commission, "NRC Inspection 05200010/2005-201 Unresolved Items," October 27, 2006. (ADAMS Accession No. ML063170069).

David B. Matthews, Nuclear Regulatory Commission, to David H. Hinds, GE Nuclear Energy, "NRC Inspection Report for General Electric Nuclear Energy (GENE) General Economic and Simplified Boiling Water Reactor (ESBWR) Quality Assurance Implementation Follow-up Inspection," January 19, 2007. (ADAMS Accession No. ML070100142).