21. TESTING AND COMPUTER CODE EVALUATION

21.1 Introduction

The General Electric (GE) 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 in NEDC-33079P, "ESBWR Test and Analysis Program Description," Class III, Revision 1, issued November 2005, 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 a letter from W.D. Beckner (NRC) to L.M. Quintana (General Electric Nuclear Energy (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," dated 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 of the *Code of Federal Regulations* (10 CFR) Section 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 for 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 on this basis. If any changes result in core uncover for any postulated LOCA, 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 few 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, noncondensable 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 test and analysis program description, 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, issued January 2000, documents this "base" qualification. This section describes specific tests related to the SBWR/ESBWR and test facilities beyond those used to create 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 noncondensable 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 gravity-driven integral full-height test for passive heat removal (GIRAFFE) and passive Nachwarmeabfuehr-und DrueckAbbau Testanlage (passive decay heat removal and depressurization test facility) (PANDA).

TRACG GDCS performance predictions were performed against the gravity-driven integrated systems test (GIST) and GIRAFFE/SIT test series. Pretest predictions have also been performed for the performance analysis and testing of heat removal system (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 before design certification. However, plant-specific hardware will be tested before or during startup testing of the plant as part of the completion of the inspection, test, analysis, and acceptance criteria (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 presented 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 Massachusetts Institute of Technology/University of California at Berkeley 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 noncondensable 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," issued March 1994.

The Single Tube Condensation Test Program was conducted to investigate steam condensation inside tubes in the presence of noncondensables. 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 used one configuration. The researchers ran tests with pure steam, steam/air, and steam/helium mixtures with representative and bounding flow rates and noncondensable 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. DBA LOCAs representing a main steamline break (MSLB), bottom drainline break (BDLB), GDCS line break, and a no-break scenario (e.g., a loss of feedwater) were simulated.

Test data, documented in GEFR-00850, "Simplified BWR Program Gravity-Driven Cooling System (GDCS) Integrated Systems Test," issued October 1989, have been used in the qualification of TRACG to the ESBWR. Tests were completed in 1988 and documented by GEH in 1989. GIST data are 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, issued November 1996, 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 noncondensable 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 are 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 Elettrica at Società 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 noncondensables 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 are 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, issued 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 noncondensable 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, issued October 1995; NEDC-33082P, Revision 1, "ESBWR Scaling Report," Class III, January 2006; and the response to Request for Additional Information (RAI) 6.3-1. 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 and RAI 6.3-1 resolution 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 earlier 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 and gave due consideration 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 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 GIRAFFE test facility included all the major components of the SBWR design and had the capability to perform 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 noncondensable gases on PCCS performance and potential systems behavior (e.g., isolation condenser (IC) operation during a 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 October 27, 1995,” dated 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 of initializing 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.

Posttest 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 responses 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 partly the result of 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. No detrimental systems interactions were apparent, 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 by using remote manual actuation of facility components. A small control room staff was required to accomplish those tasks. The staff followed the written procedures closely, and steps were noted on a test log/checklist, which was signed by the test engineer. Toshiba staff operated professionally, and appropriate consideration of testing QA appeared 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 constitute well-run test programs 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 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 MFN 170-94, "Summary of the visit on October 16, 1994, at the Società Informazioni Esperienze Termoidrauliche (SIET) Performance Analysis and Testing of Heat Removal System (PANTHERS) Test Facility for the SBWR Design," dated December 21, 1994. Major observations from the visit are discussed below.

Testing in PANTHERS provided considerable data on PCCS heat exchanger performance. Both GEH and ENEA (which was a partner in ownership of SIET Laboratories) supervised the testing, which 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 noncondensable 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. The staff had some concerns regarding the ICS structural integrity and design, particularly the leakage in the ICS during testing at the PANTHER-IC facility. The 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

GEH and PSI performed PANDA testing as a joint effort 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 or 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 requirements appear in 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 noncondensables. The work was independently conducted at the UCB and MIT, as described in 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 used one configuration. Tests were run with pure steam, steam-air, and steam-helium mixtures with representative and bounding flow rates and noncondensable mass fractions. The results demonstrated that the system behaved as expected for all tests, with either of the noncondensables, 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 the PCCS primary side is a function of the mass flow rate and noncondensable 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 noncondensables allows the PCCS heat removal rate to match the reactor decay heat for the long term. The staff, therefore, finds that the test data are sufficient for developing the condensation heat transfer correlation used in the TRACG code and meet 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 noncondensable 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.

GEH, Ansaldo Spa, ENEA, and SIET performed PANTHERS/PCCS testing as a joint effort 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 noncondensable 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 noncondensable gases, poolside heat transfer, parallel PCCS tube effects, and parallel PCCS module 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 (to simulate nitrogen in the containment), steamflow rates, operating pressures, and superheat conditions that cover the expected ranges of values of the parameters for the ESBWR.

Many 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. No measurements were taken of mass flow rate and noncondensable 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. Sections 21.3.1.1 and 21.5.3.1 of this report discuss the UCB test program.

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 noncondensable 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 noncondensable 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 noncondensable 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 driven by natural circulation, 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 noncondensable 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, effect of the pool's water level 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 of 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. No measurements were made of the mass flow rate and noncondensable 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 such a correlation 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.

The 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, DCD, Tier 2, Table 14.2-1, 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 the ICS. Therefore, the staff 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 affect 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. Hence, the staff finds that testing at high power would not be more challenging from the viewpoint of the structural integrity of the ICS, and no DCD change is required. Based on the applicant's response, RAI 14.2-3 was resolved.

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 Tests

Researchers conducted full-size depressurization valve (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 that GEH submit the test reports for the DPVs. Section 3.9 of this report presents the staff's evaluation of the DPV test results. Based on the applicant's response, RAI 3.9-1 was resolved.

Vacuum Breaker Tests

Researchers conducted full-size VB tests at a facility in Italy. The test objective was to demonstrate reliable operation of the VB. The opening and closing pressures of a VB were measured.

In RAI 3.9-1, the staff also requested that GEH submit the VB test reports. Section 3.9 of this report provides the staff's evaluation of the VB test results. Based on the applicant's response, RAI 3.9-1 was resolved.

21.5.3.2 Integral Systems Tests

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

GIST 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.

GIST focused on the ability of the GDCS to maintain core cooling in a LOCA. GEH performed the tests in San Jose, CA, 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 the real-time response of the 1987 SBWR pressures and temperatures.

The GIST 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 1,210 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 PANDA and GIRAFFE design 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 noncondensable gas, including the process of purging noncondensable 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 noncondensable 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 noncondensable gas (using helium as a substitute for hydrogen gas) and a heavier-than-steam noncondensable 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 pounds-force per square inch absolute). 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 noncondensable 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. However, 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 considered in the code uncertainty evaluation.

Only two noncondensable gas sampling locations were 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 noncondensable gas behavior in the drywell. This problem was compounded by the scarcity of the noncondensable sampling locations. For the wetwell gas space, there was only one noncondensable gas sampling location. However, unlike the lower drywell, the wetwell wall heat loss was found to be insignificant. The scarcity of the noncondensable gas sampling locations and the heat loss problem at the lower drywell tended to reduce the quality of the containment noncondensable 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) noncondensable gases present in the containment under various test conditions.

Because of the heat loss at the lower drywell, noncondensable gas distribution in the drywell is distorted by having a much higher noncondensable concentration (because of local steam condensation) than expected in the lower drywell. Furthermore, since there were only two noncondensable 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 noncondensable 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 noncondensable 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 noncondensable 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 considered in 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 asked 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, feedwater line break (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 volume in the lower drywell.

GEH further stated that Table 2 in MFN 05-109, "GE Response to Results of NRC Acceptance Review for ESBWR Design Certification Application—Item 2," dated 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" (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 that the GEH response resolves RAI 21.5-1, 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 1,040 seconds (transition from the GDCS injection phase to the long-term cooling phase) to about 3,600 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.

The 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 1,040 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 the previous discussion of RAI 21.5-1.)

The volume of the GDCS pool was much smaller than the scaled volume, and consequently, the amount of water was insufficient 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, which 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 the previous discussion of 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. However, 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 that GEH 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 boiloff 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 noncondensable 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 noncondensable 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 the 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 in 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 noncondensable 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 noncondensable 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 the previous discussion of 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 noncondensable 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 noncondensable gas injection to the drywell during a test.

On the basis of the preceding discussion, the staff concludes that, despite the 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, the tests provided data on PCCS performance with noncondensable 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 finds 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 scaleup 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. 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," issued 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, issued December 2002, 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 in more detail at specific processes important to system behavior. For the ESBWR, the analysis identified 46 highly ranked phenomena needing detailed evaluation and providing 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 states 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 represents 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 the process that 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 suggests 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, that "these equations are applied to the specific regions of the ESBWR" raised the question of whether the GEH original scaling approach ignored interactions. Even when two or three volumes were 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.

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 behave similarly. 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 that GEH submit additional information to address the concerns identified above.

GDCS Transition Phase

In RAI 6.3-1, the staff asked GEH to perform a revised scaling analysis for the 4,500-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 to RAI 6.3-1, GEH performed an updated ESBWR scaling analysis. 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 the article by M. di Marzo, “A Simplified Model of the BWR Depressurization Transient,” Nuclear Engineering and Design, 205 (2001), pp. 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 4,000-MWt ESBWR, it also used the same GDCS injection line break for the 4,500-MWt ESBWR as an example in the updated analysis to allow a comparison of the results.

GEH also presented the results for the 4,500-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 that from 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 4,500-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 4,500-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 few data on multidimensional phenomena were obtained. 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 explain 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 noncondensable, 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 noncondensable 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 heatup 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 noncondensables to the wetwell. As a consequence, GEH concluded that, by considering transfer of all drywell noncondensables 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 noncondensables 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 is required for the long-term cooling phase. Based on the applicant's response as described in the preceding paragraphs, RAI 6.3-1 was resolved.

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. GEH 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, noncondensable 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 scaleup 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 meets the requirements delineated in 10 CFR 40.43(e) as referenced by 10 CFR 52.47(b)(2)(c)(2) and discussed below.

ESBWR plant features appeared in earlier BWR designs which have provided satisfactory operation over many 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 the ESBWR (i.e., ICS, GDCS, and PCCS). For features that are not unique to the 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 data from operating BWRs over a wide range of reactor conditions, including temperatures and pressure conditions in 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 is described in Sections 2 and 3 of NEDC-33083P-A, MFN 05-017, "TRACG Application for ESBWR," March 2005.

- Stability analysis is described in NEDC-33083P, Supplement 1, Revision 1, "TRACG Application for ESBWR Stability," January 2008.

- ATWS analysis is described in NEDE-33083P, Supplement 2, Revision 2, "TRACG Application for ESBWR Anticipated Transient Without Scram Analyses," September 2009.

- Non-LOCA transients, including anticipated operational occurrences (AOOs) and infrequent events (IEs), are described in NEDC-33083, Supplement 3, Revision 1, "TRACG Application for ESBWR Transient Analysis," September 2009.

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 refers to the 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 for NEDC-33083P, Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design," dated January 11, 2008, addresses the evaluation of the 20 confirmatory items.

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. The NRC letter "Reissuance of Safety Evaluation Regarding the Application of the GE-Hitachi Nuclear Energy Americas LLC (GEH) LTR 'TRACG Application for the ESBWR Stability Analysis,' NEDE-33083P, Supplement 1," dated August 29, 2007, 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" addresses the evaluation of the seven confirmatory items.

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" documents the staff's evaluation. Subsequently, GEH submitted NEDC-33083, Supplement 3, to document the TRACG application to AOOs for ESBWR. The open items from the staff's review of NEDC-33083P-A Section 4, "Transient Analysis," are addressed in detail and closed, and the updated information in NEDC-33083, Supplement 3, is addressed in the Safety Evaluation Report for Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design.

The NRC staff reviewed NEDE-33083P, Supplement 2, and documented its evaluation in the "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." The open items from the staff's safety evaluation are addressed in detail and closed in the Addendum to the Safety Evaluation Report for Application of the TRACG Computer Code Anticipated Transients Without Scram for the ESBWR Design, NEDE-33083P, Supplement 2.

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 DCD.

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, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors"; 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 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," referred to hereafter as the standard review plan (SRP), Section 15.0.2, "Review of Transient Accident and Analysis Methods," issued December 2005.

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 <u>Documentation</u>

The development of an evaluation model for use in reactor safety licensing calculations requires substantial documentation.

SRP Section 15.0.2 requires that this documentation 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:

- evaluation model description and theory manual (NEDE-32176P, Revision 4, "TRACG Model Description," dated January 31, 2008)

- licensing topical reports (LTRs) that cover accident scenario identification and uncertainty analysis

- NEDC-33083P-A for LOCA
- NEDC-33083P, Supplement 1 for stability
- NEDE-33083P, Supplement 2 for ATWS
- NEDE-33083P, Supplement 3 for AOO/IE

- code assessment

  - NEDE-32177, Revision 3, “TRACG Qualification,” dated August 29, 2007

  - MFN 04-059, “Update of ESBWR TRACG Qualification for NEDC-32725P and NEDC-33080P Using the 9-Apr-2004 Program Library Version of TRACG04,” dated June 6, 2004

  - NEDC-32725P, Revision 1, “TRACG Qualification for SBWR,” dated August 30, 2002

  - NEDC-33080P, “TRACG Qualification for ESBWR,” dated June 2, 2004

- user’s manual—UM-0136, Revision 0, “TRACG04A, P User’s Manual,” issued December 2005

- QA program (see Section 21.6.2.6 of this report for discussion of the QA program)

21.6.2.2 <u>Evaluation Model</u>

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 <u>Accident Scenario Identification Process</u>

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-33083P
LOCA—long-term core cooling		MFN 05-105
AOO/IE	2.3-3	NEDC-33083P, Supplement 3, Revision 1
Stability	2.3-5	NEDC-33083, Supplement 1, Revision 1
ATWS	2.3-4	NEDC-33083, Supplement 2, Revision 2

21.6.2.4 <u>Code Assessment</u>

The code assessment provides a complete assessment of all code models compared to 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 the following:

- NEDE-32177P, Revision 3, "TRACG Qualification," issued August 2007

- MFN 04-059, "Update of ESBWR TRACG Qualification for NEDC-32725P and NEDC-33080P Using the 9-Apr-2004 Program Library Version of TRACG04," dated June 6, 2004

- NEDC-32725P, Revision 1, "TRACG Qualification for SBWR," dated August 30, 2002

- NEDC-33080P, "TRACG Qualification for ESBWR," dated June 2, 2004

21.6.2.5 <u>Uncertainty Analysis</u>

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 (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 performed the calculation using bounding assumptions instead.

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, "Domestic Licensing of Production and Utilization Facilities." Staff performed two audits of GEH's QA plan. The first audit took place between October 16 and October 19, 2006, resuming between October 30 and November 3, 2006 (referred to as the October 2006 audit). The second audit took place between December 11 and December 15, 2006, resuming between December 19 and December 20, 2006 (referred to as the December 2006 audit). The third audit took place between December 15 and December 19, 2008.

GEH has procedures that meet 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 GEH staff can then use 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. "Level 2R" is a specific status defined by GEH procedure in the Level 2 process, which may be applied to design tasks for a limited time. The QA approving official must still approve Level 2R codes.

At the time the 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, the NRC staff requested that GEH inform the staff placing the TRACG04 code under a QA-approved code change control process and provide information to the NRC staff sufficient for its use in reviewing and approving the version of TRACG04 used to develop the final ESBWR DCD submittal. The staff was tracking RAI 21.6-109 as an open item in the SER with open items.

On December 15–19, 2008, the NRC conducted an inspection at the GEH facility in Wilmington, NC. The inspection assessed GEH's compliance with selected portions of Appendix B to 10 CFR Part 50 and the provisions of 10 CFR Part 21, "Reporting of Defects and Noncompliance." During the inspection, the NRC inspectors found that implementation of GEH's QA program failed to document the justification for the use of a particular version of a non-Level-2 code during alternate calculations to verify original calculations and assumptions. Specifically, the NRC staff noted that Appendix F to Global Nuclear Fuel Common Procedure 03-09, "Independent Design Verification," Revision 1, dated January 4, 2006, allows for use of several different versions of non-Level-2 code with appropriate justification. During the inspection, the NRC staff did not find evidence that the use was justified, as there was no documentation concerning when non-Level-2 codes were used. At the end of the inspection, GEH prepared a Corrective Action Request (CAR 47253) to address this issue. This issue was identified as Notice of Violation (NOV) 05200010/2008-201-02 in Inspection Report 05200010/2008-201, dated March 25, 2009.

On April 23, 2009, GEH responded to NOV 05200010/2008-201-02 by stating that it had updated ESI 30-01, “Alternate Calculations for Verification of Non-Level-2 Computer Code Calculations,” on March 23, 2009, to require written justification for use of a non-Level-2 version of the TRACG04 code when applied to all GEH and Global Nuclear Fuel analysis activities. In its response, GEH also committed to the performance of safety analyses for the ESBWR DCD, Revision 6, and future revisions of LTRs NEDO-33337, “ESBWR Initial Core Transient and Accident Analysis,” and NEDO-33338, “ESBWR Feedwater Temperature Operating Domain Transient and Accident Analyses,” with the TRACG04P Level 2 ECP. As a result, non-Level-2 versions of TRACG will not be applied to future DCD analyses, alleviating the need for further discussion of the rationale concerning code versions. Based on the information in the GEH response, the staff determined that RAI 21.6-109 and NOV 05200010/2008-01-02 were resolved.

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). 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 to Version 52 (i.e., the Level 2 version used for all ESBWR design certification calculations).

Although GEH performed part of the TRACG04 assessment with a different version of the code than the version 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 had 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 overall the changes were insignificant 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, Revision 4, “Quantifying Reactor Safety Margins: Application of Code Scaling Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident,” issued December 1989). 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.

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. On July 31, 2008, and January 9, 2009, the staff received GEH's response to RAI 21.6-92 and the associated Supplement 1, respectively, which provided detailed tables with transient cases, code version used, QA level, hardware/operation system, and executable revision date. On March 16, 2009, GEH provided the staff with TRACG versions used for feedwater temperature operation domain and initial core topical report analysis. Based on the satisfactory response to the staff's request for exact code revisions, version numbers, and TRACG versions used in applicable LTRs, RAI 21.6-92 was resolved.

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 analyses
- non-LOCA transients, including AOOs and IEs
- stability analysis
- ATWS analysis

The references listed below 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 are proprietary and will not be discussed in detail in this document; however, the following references document these bases:

- NEDC-33083P-A, MFN 05-017, "TRACG Application for ESBWR," issued March 2005

- Safety Evaluation Report Regarding the Application of General Electric's Topical Report, TRACG Application for ESBWR Stability Analysis, NEDE-33083P, Supplement 1, dated August 29, 2007

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

- Addendum to the Safety Evaluation Report for NEDC-33083P, Supplement 1, "TRACG Application for ESBWR Stability," dated October 29, 2009

- Office of New Reactors, 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

- Safety Evaluation Report for "Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design," NEDE-33083P, Supplement 3

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, Supplement 3), stability (NEDC-33083P, Supplement 1,), and ATWS (NEDE-33083P, Supplement 2) provide an overview of the respective evaluation models that describe all parts of the evaluation model, the relationships between them, and where they are located in the documentation. These LTRs 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 RPV water level for the long-term core cooling phase of the LOCA. The staff received information on ESBWR long-term core cooling in letter MFN 05-105, from D.H. Hinds (GE) to the NRC, "TRACG LOCA SER Confirmatory Items (TAC No. 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," dated October 6, 2005.

The topical reports also contain a determination of the code uncertainty for a sample plant calculation. In NEDC-33083P-A, GEH demonstrates the bounding LOCA calculation, instead of determining an uncertainty for this event. It is acceptable to use a bounding TRACG LOCA calculation for the ESBWR because the substantial margin prevents the ESBWR core from uncovering during a LOCA. Section 21.6.3.5.1 contains further discussion of this issue.

In NEDE-32177P (Revision 2), MFN 04-059, and NEDC-32725P (Revision 1), GEH provided the code assessment for TRACG. These documents include a description of each assessment test, the reason it was chosen, acceptance 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 that GEH provide an update to the TRACG qualification report (NEDE-32177P, Revision 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 (NEDE-32177) on August 29, 2007. In NEDE-32177, Revision 3, qualification cases have been added in each of the major qualification categories. In the separate effects category, the additional cases (1) extend the range of the void fraction qualification to lower pressures and larger diameter geometries, (2) provide a qualification basis for TRACG prediction of core spray heat transfer, and (3) extend the fuel bundle pressure drop and critical power qualifications to include the current 10×10 fuel design. In the component category, a set of qualification cases has been added to evaluate the capability of the TRACG mechanistic core spray distribution model. In the integral systems category, qualification studies using data from the ROSA, FIX, and GIST test facilities have been added to provide additional support for TRACG LOCA applications. Finally, in the BWR plant category, qualification studies using stability data from Peach Bottom and Nine Mile Point have been added to support the application of TRACG for prediction of plant stability. The NRC staff determined that the updated information is sufficient to extend the qualification to TRACG04. Therefore, based on the applicant's response, RAI 21.6-75 was resolved.

The staff determined that NEDE-32176P, Revision 4, 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, Revision 0) provides detailed instructions about how the computer code is used; a description of how to choose model input parameters and appropriate code options; guidance on 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; NEDE-33083P, Supplement 2; and NEDE-33083P, Supplement 3) provide additional guidance on specific modeling of the events.

During three audits of GEH records, the staff reviewed the GEH documentation for the 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 (4,000-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 these changes would not alter the staff's conclusions. In the response to RAI 21.6-98 received on August 29, 2008, GEH provided the design changes and TRACG model justifications. In the response to RAI 21.6-98, Supplement 1, which the NRC received on March 3, 2009, GEH agreed to document the design changes and TRACG model justification in LTR NEDE-33440P, "ESBWR Safety Analysis—Additional Information," Revision 1 (issued June 2009). In the response, GEH listed all design changes that have impacts on the LOCA analysis since the approval of TRACG for ESBWR LOCA analysis (NEDC-33083P-A) through ESBWR DCD, Tier 2, Revision 5. With these new updates, the LOCA has been reanalyzed and documented in Sections 6.2 and 6.3 in the ESBWR DCD, Revision 6. Sections 6.2 and 6.3 of this report document the staff's evaluation of the LOCA analysis. The staff finds that Confirmatory Items 14 and 20, as mentioned in RAI 21.6-98, were addressed sufficiently. Based on the applicant's response, RAI 21.6-98 was resolved.

Because GEH provided some of the appropriate updates to the documentation in RAI responses, in RAI 21.6-63, Supplement 1, and in RAI 21.6-65, Supplement 2, the staff requested that GEH submit the 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 stand-alone new topical report or a new supplement to NEDC-33083. On February 15, 2008, the staff received the response to RAI 21.6-63, Supplement 1, and RAI 21.6-65, Supplement 2, in which GEH stated that the requested information was consolidated in NEDE-33083, Supplement 3, "TRACG Application for ESBWR Transient Analysis", issued December 2007. A separate SER for NEDC-33083P, Supplement 3, provides the staff's technical evaluation of this LTR. Therefore, based on the applicant's response, RAIs 21.6-63 and 21.6-65 were resolved.

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 noncondensable 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. From the comparisons documented in NEDE-32177P, TRACG demonstrates 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. Therefore, 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 the 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 noncondensables. 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 include 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 demonstrate the acceptability of the TRACG capability to predict wall heat transfer.

21.6.3.2.4 Post-Critical 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 the ESBWR experiences post-CHF heat transfer.

The NRC staff reviewed the applicability of this model for ESBWR ATWS events as discussed in the Addendum to the Safety Evaluation for Application of the TRACG Computer Code to Anticipated Transients Without Scram for the ESBWR Design NEDE-33083P, Supplement 2.

21.6.3.2.5 Flow Regime Maps

A two-fluid formulation relies on 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 uses 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. The staff reviewed the flow regime maps and transition between flow regimes and found them acceptable for the stated ESBWR applications. NEDC-33083P-A and the staff's Addendum to the Safety Evaluation Report for NEDC-33083P-A, "Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design," document the details of the evaluation.

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 expected in the reactor. The code uses a critical Weber number criterion for estimating interfacial area density or bubble/droplet diameter. However, the way this approach is used differs for interfacial momentum and heat transfer in bubbly flow and droplet flow. NEDE-32176P, Revision 2, “TRACG Model Description,” issued December 1999, 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 the 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 varied pressure, flow rate, and inlet subcooling. 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 requested in RAI 21.6-75 that GEH submit the updated qualification report (NEDE-32177). In response, GEH submitted Revision 3 of the TRACG qualification report on August 29, 2007. Based on this submittal, RAI 21.6-75 was resolved.

The staff reviewed the GEH qualification of its void fraction data provided in this report to ensure that the modifications to the entrainment fraction and its subsequent use in the interfacial shear model compare well with data. The void fraction assessment results from NEDE-32177P, Revision 3, are very close to the results from NEDE-32177P, Revision 2, which was assessed as acceptable during the preapplication phase of the ESBWR design certification review. This ensures that the conclusion from the preapplication TRACG review is still valid. In addition, NEDE-32177P, Revision 3, adds assessment cases, which include Toshiba Low-Pressure Void Fraction Tests, Ontario Hydro Void Fraction Tests, and Centro Informazioni Studi Esperienze, SpA (CISE) Density Measurement Tests. The Toshiba tests were added to extend the qualification basis to lower pressures at 0.5 and 1.00 MPa. The Ontario Hydro facility provides the void fraction database for a large-scale pumped flow facility. The CISE facility in Italy provided data concerning the void and quality relationship. The TRACG assessment showed reasonable agreement with the data from those tests. The assessment from NEDE-32177P, Revision 3, reinforced the conclusion from the approved GEH LTR NEDC-33083P-A that the interfacial shear model is acceptable.

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, Revision 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 noncondensable 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, 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 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.

In an RAI (6.3-13) related to the choked flow model used in ESBWR LOCA analyses, the staff asked GEH to include the RPV injection line nozzle and equalizing line nozzle throat length to diameter (L/D) ratios in ITAAC to ensure that the L/D ratio remains within the applicability range of the TRACG code flow choking model for LOCA calculations. In NEDE-32176P, Revision 3, Section 6.3 describes the TRACG04 choked flow model, and Section 6.3.3 describes the calculation of the sonic velocity. In Section 6.3.3, GEH stated the simplifying assumptions used to calculate the sonic velocity, which include assuming equilibrium conditions. GEH stated, "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. By letter dated March 5, 2008, the applicant submitted its response to RAI 6.3-13, Supplement 1. The response included the following information:

- TRACG has a subcooled choking model applicable for small L/D throat conditions. The model prediction comparisons to data include choked flow for both smooth and abrupt area changes (i.e., orifices), thus validating the model for small L/D conditions.

- TRACG is qualified over a range of L/D of 0.0–8.68 through direct comparison to test data. GEH provided a table of tests, which contains the L/D for the PSTF critical flow tests, Marviken, and the Edwards Pipe Tests used to qualify the TRACG critical flow model.

- GEH provided a table of L/Ds for ESBWR break lines. The values of L/Ds of ESBWR break lines are within the ranges of the TRACG qualification database.

Recognizing that it is not possible to have continuous L/D values in the range of test data, the ESBWR break throat L/D values are within the range of tests used to qualify the TRACG code choking model. Based on the RAI response, the staff concluded that the TRACG model covers L/Ds for the ESBWR break lines. The staff reiterated the original request for an ITAAC on throat L/Ds in RAI 6.3-13, Supplement 2. In response to RAI 6.3-13, Supplement 2, GEH added ITAAC 13 and 14 to the GDCS Table 2.4.2-3 of Tier 1 of the ESBWR DCD. Therefore, based on the applicant’s response, RAI 6.3-13 was resolved.

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 the 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 code predictions that fall 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 on 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.

In addition, as is the case with all thermal-hydraulic system codes today, there is an inconsistency in the interfacial area used for momentum transfer and heat transfer in bubbly and droplet flows. However, the staff finds that the good agreement with the void fraction and heat transfer data shows that these limits do not adversely affect the results.

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. GEH added the option of using the Shumway correlation to calculate the minimum stable film boiling temperature in TRACG, 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 supports the continued use of the Iloeje correlation for ESBWR applications for LOCA, AOO, and ATWS. For LOCA and AOO events for the ESBWR, the core does not enter film boiling, and therefore this correlation is not used. For ESBWR ATWS events in which the core does go into film boiling, the minimum stable film boiling temperature is used only to determine when the core will quench. In a supplement to RAI 21.6-79, the staff asked GEH to justify the high ranking of this parameter 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," dated February 28, 2006.) The GEH response to RAI 21.6-79, Supplement 1, explained that PIRT C13 pertains to both dryout and rewet/boiling transition. C13 is ranked as a high PIRT parameter for both ESBWR and BWR/2–6 AOO events because of the importance of calculating a margin to dryout. However, no ESBWR or BWR/2–6 AOO events exceed the maximum critical power ratio (MCPR) where minimum stable film boiling temperature may be encountered. Therefore, the rewet portion of C13 is not of high importance. The staff accepts GEH's explanation as to why C13 is ranked as a high PIRT parameter for both ESBWR and BWR/2–6 AOO events. Based on the applicant's response, RAI 21.6-79 was resolved.

21.6.3.2.12 Critical Heat Flux

For CHF, TRACG has a proprietary correlation, the GE critical quality boiling length correlation (GEXL), which is based on the critical quality concept for normal flows and uses a modified Zuber correlation for low flows and flow reversal. Section 4.4 of this report discusses the applicability of the GEXL correlation for the ESBWR and the RAIs associated with this topic and finds the use of the GEXL correlation for the ESBWR to be acceptable.

21.6.3.2.13 Gap Conductance

TRACG has the option of using either a constant or dynamic gap conductance. The dynamic gap conductance is used in the 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 a 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 NEDC-33083P-A 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 stoichiometric mixture of steam and hydrogen from the metal-water reaction. The constants in the equation for the gas conductivity in a perforated fuel rod are TRACG input constants. The ESBWR is not expected to experience rod perforation during any LOCA, stability, AOO/IE, or ATWS event. NEDC-33083P-A documents this model. 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 of the staff’s review of the TRACG gap conductance model.

21.6.3.2.14 Fuel Rod Thermal Conductivity

The default fuel thermal conductivity modeling in TRACG04 is based on the PRIME03 code, which the NRC has not reviewed and approved for the ESBWR. RAI 6.3-54 requested that GEH justify use of the PRIME03-based thermal conductivity model in TRACG04, since PRIME03 has not been reviewed and approved by the NRC for the ESBWR. RAI 6.3-55 requested that GEH justify the use of gap conductance and fuel thermal conductivity from different models (GSTRM and PRIME03-based TRACG04, respectively).

GEH’s response to RAI 6.3-55 included a description of the TRACG04 calculations, as discussed in the following paragraphs concerning RAI 6.3-54. The response to RAI 6.3-55 did not provide sufficient justification for combining models. However, the response to RAI 6.3-54, Supplement 1, addressed the impact of using gap conductance and fuel thermal conductivity from different models (GSTRM and PRIME03-based TRACG04, respectively) on TRACG04 calculations. Since the supplements to RAI 6.3-54 addressed the issue, the staff concluded that RAI 6.3-55 could be closed.

The GEH response to RAI 6.3-54 states that the fuel files generated using the GSTRM code are being used as input to TRACG04 and that the TRACG04 thermal conductivity model is used. The TRACG04 thermal conductivity model is based on the thermal conductivity model in the PRIME03 code and accounts for the degradation of thermal conductivity due to the presence of gadolinium and for the degradation of thermal conductivity as exposure increases. Since the TRACG04 thermal conductivity model has not been approved in previous versions of TRACG and since the thermal conductivity model has not been approved as part of a PRIME03 review for the ESBWR, in RAI 6.3-54, Supplement 1, the NRC staff requested that GEH provide experimental data and benchmarks, as well as study results comparing TRACG02 (GSTRM) to TRACG04 (PRIME03-based) thermal conductivity sensitivity. In responding to RAI 6.3-54, Supplement 1, in MFN 08-713, GEH provided the results from sensitivity studies comparing representative AOO, ATWS, and stability cases analyzed with the GSTRM model and the TRACG04 (PRIME03-based) model to the base cases using GSTRM gap conductance and TRACG04 (PRIME03-based) thermal conductivity. GEH did not submit experimental data and benchmarks to support use of the PRIME03 code or the TRACG04 thermal conductivity model for the ESBWR.

For AOOs, the generator load rejection with total bypass failure (LRNBP) from ESBWR DCD, Section 15.3.5 was selected as the transient event (AOO/IE) for the sensitivity study. This transient event is expected to be the one most affected by the fuel thermal conductivity and gap conductance because it is a fast event with the most severe flux peak. The LRNBP sensitivity study results in the response to RAI 6.3-54, Supplement 1 (Table 6.3-54-1), show negligible differences (< 1% ΔP and < 0.005 ΔCPR/ICPR) in maximum dome pressure, maximum vessel bottom pressure, and ΔCPR/ICPR. In addition, RAI 6.3-54, Supplement 2, requested that GEH provide the fuel centerline temperatures and melting temperatures for the LRNBP cases.

The results transmitted in the response to RAI 6.3-54, Supplement 2 (MFN 08-713, Supplement 1) show that the base LRNBP case (GSTRM gap conductance and PRIME03-based thermal conductivity) yields the most conservative maximum fuel centerline temperature. The response to RAI 6.3-54, Supplement 2, also shows that the uranium dioxide melting temperatures for all three cases are identical, since melting temperature is a function of exposure, and all of the cases assume the same exposure.

The main steam isolation valve (MSIV) closure from ESBWR DCD, Section 15.5.2 was selected as the ATWS event analyzed in the sensitivity study. This ATWS event is expected to be the one most affected by the fuel thermal conductivity and gap conductance. The MSIV closure sensitivity study results shown in the response to RAI 6.3-54, Supplement 1 (Table 6.3-54-2), show negligible differences in associated containment pressure (approximately 1% ΔP), maximum bulk SP temperature (< 2 degrees Fahrenheit (F)), and peak cladding temperature (< 10 degrees F).

The AOO and ATWS sensitivity study results documented in the response to RAI 6.3-54, Supplement 1, provide the staff with reasonable assurance that the transient event analyses shown in the ESBWR DCD and in the TRACG analyses for ESBWR AOO and ATWS topical reports do not exceed acceptance criteria. Based on the applicant’s response, RAI 6.3-54 was resolved.

The staff accepts the use of the GSTRM model for both gap conductance and thermal conductivity in the ESBWR design certification. The conclusions and limitations for ESBWR TRACG AOO and ATWS analyses (including the 350-psi critical pressure penalty) contained in the NRC staff evaluation of GEH’s Part 21 report (Appendix F to the safety evaluation for NEDC-33173P) are applicable to this safety evaluation. The NRC must approve the use of other methods or analysis strategies for the ESBWR.

The loss of feedwater heating (LOFWH) regional stability evaluation at middle-of-cycle exposure from ESBWR DCD, Section 4D.1.5 was chosen for the stability sensitivity study because it is the limiting stability event. A comparison of the decay ratio results from the base case and two sensitivity cases shows a negligible change in decay ratio between the base case (with GSTRM gap conductance and PRIME03-based thermal conductivity) and the PRIME03-based sensitivity case. However, the GSTRM sensitivity case resulted in a relatively limiting decay ratio (0.71 for the GSTRM case versus 0.66 for base case and PRIME03-based case).

The results of the LOFWH regional stability sensitivity studies give the staff reasonable assurance that the ESBWR TRACG 0.8 decay ratio limit is not exceeded by the stability results shown in the ESBWR DCD and in TRACG for the ESBWR stability topical report. In addition, the stability analysis will be analyzed on a cycle-specific basis, so these results will be updated.

The staff accepts the use of the GSTRM model for both gap conductance and thermal conductivity in the ESBWR design certification. The conclusions and limitations for ESBWR stability analyses (including the 350-psi critical pressure penalty) contained in the NRC staff evaluation of GEH’s Part 21 report (Appendix F to the safety evaluation for NEDC-33173P) apply to this safety evaluation. The NRC must approve the use of other methods or analysis strategies for the ESBWR.

GEH did not include LOCA sensitivity studies in response to RAI 6.3-54, Supplement 1, because the water level remains above the top of active fuel in ESBWR LOCA analyses. Consequently, there is no fuel heatup. The impact of fuel thermal conductivity and gap conductance is much less significant than in cases where fuel heatup is calculated. In addition, dynamic gap conductance is not used in LOCA analysis because the PIRT parameters related to gap conductance were not determined to be of high importance to ESBWR LOCA analysis (NEDC-33083). The NRC staff performed confirmatory calculations of ESBWR LOCA fuel conductivity sensitivity using the TRACE model. The results showed that the minimum water level in the limiting LOCA is not sensitive to the 30-percent fuel thermal conductivity reduction. (The 30-percent fuel thermal conductivity impact was a bounding reduction used by the staff in its confirmatory calculations to verify the GEH calculation results showing that AOO and IE results are not sensitive to the PRIME and GESTR fuel thermal conductivity model differences.)

Therefore, the staff has reasonable assurance that the LOCA acceptance criteria are not exceeded in the LOCA analyses in the ESBWR DCD and in the TRACG for ESBWR LOCA analysis topical report.

The staff accepts the use of the GSTRM model for both gap conductance and thermal conductivity in the ESBWR design certification. The conclusions and limitations for ESBWR TRACG LOCA analyses (including the 350-psi critical pressure penalty) contained in the NRC staff evaluation of GEH's Part 21 report (Appendix F to the safety evaluation for NEDC-33173P) are applicable to this safety evaluation. The NRC must approve the use of other methods or analysis strategies for the ESBWR.

21.6.3.2.15 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 cross sections generated by PANAC11/TGBLA06 as input by means of a PANAC11 "wrapup" file. GEH submitted the contents of the wrapup file, which the staff reviewed 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. The staff's "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)" includes the staff's review of these processes.

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. In RAI 4.4-2 and the associated supplements, 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. The staff's safety evaluation in Section 4.4 of this report discusses the resolution of RAI 4.4-2 and the associated supplements.

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

21.6.3.2.15.1 Void Coefficient

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 NRC staff requested that GEH provide the documentation of the evaluation of the TGBLA06 lattice calculations relative to MCNP in order to reestablish the void coefficient correction model as applied to TRACG.

The GEH response to RAI 21.6-84 notes that the void coefficient reassessment was completed for TRACG04 in topical report NEDE-32906P, Supplement 3, issued May 2006. This reassessment for TGBLA06 against MCNP results occurred, and the void coefficient correlation model to be applied for TRACG04 was updated prior to performing the TRACG04 qualification cases that required the use of the three-dimensional neutron kinetics model. A detailed discussion of this item appears in Section 3.20.2 of the staff's "Final Safety Evaluation of GEH's LTR NEDE-32906P, Supplement 3, 'Migration of TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,'" dated July 10, 2009. Based on the applicant's response, RAI 21.6-84 was resolved.

21.6.3.2.15.2 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 LOFWH 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 MSIV closure event. In addition, although GEH ranks this parameter as having "medium" importance, it still includes the uncertainty in the uncertainty analysis for the ESBWR AOO and IE. The staff recognizes that the uncertainty for the Doppler coefficient was determined based on PANAC10. Based on the applicant's response RAI 21.6-60 was resolved. 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," issued May 2006), the staff requested that GEH justify that the uncertainty is applicable or bounding for PANAC11.

RAI 21.6-108, in regard to NEDC-33083P-A, also addressed this question. In that RAI, the NRC staff requested that GEH demonstrate that PIRT items C1BX and C1CX (uncertainty in the

Doppler coefficient and in scram reactivity, respectively) in Table 4.4-1 of NEDC-33083P-A are still applicable or bounding when applying the PANAC11 physics methods.

GEH responded to RAI 21.6-108 in MFN 08-598, which stated that GEH had previously addressed this issue in its response to NEDC-32906P RAI 21. That response, transmitted in MFN 08-547, is summarized as follows:

- The scram reactivity uncertainty (C1CX) is dominated by the uncertainty in determining the scram speed. The scram speed uncertainty is determined based on plant data obtained from scram speed tests at BWR plants and does not depend on the lattice or core physics methods.

- The Doppler coefficient uncertainty (C1BX) is conservatively determined based on calculated responses for the SPERT tests. C1BX was not directly established from lattice physics calculations. Therefore, this parameter does not directly depend on the lattice physics methods.

MFN 08-547 contains GEH analysis results that show the continued applicability of the Doppler coefficient uncertainty to the TRACG04 AOO calculations.

Based on the GEH responses in MFN 08-598 and MFN 08-547, RAI 21.6-108 was closed.

21.6.3.2.15.3 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, established using the PANAC10 model, are also being used for the TRACG04/PANAC11 application for the ESBWR AOO and IE.

RAI 21.6-108 addressed this question with regard to NEDC-33083P-A. In RAI 21.6-108, the NRC staff requested that GEH demonstrate that PIRT items C1BX and C1CX (uncertainty in the Doppler coefficient and in scram reactivity, respectively) found in Table 4.4-1 of NEDC-33083P-A are still applicable or bounding when applying the PANAC11 physics methods.

As discussed in Section 21.6.3.2.15.2 of this report, and based on the applicant’s response, RAI 21.6-108 was resolved.

21.6.3.2.15.4 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 GEH validation for TGBLA available for staff review is a code-to-code comparison to PANAC11.

The staff performed 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. The staff calculations confirm the validity of the TRACG boron reactivity model.

21.6.3.2.15.5 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 RAI 21.6-82, the staff requested that GEH provide information on whether transient xenon conditions were modeled in the AOO or ATWS analyses. 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 staff questioned the startup evolution, which will take place over the course of hours. In RAI 21.6-82, Supplement 1, the staff requested that GEH justify use of the constant xenon assumption for the startup simulation. GEH responded to RAI 21.6-82, Supplement 1, in a letter dated June 30, 2008 (MFN 08-457). The responses to RAIs 4.4-59 and 4.4-60 provide additional information on the xenon assumption for startup. The detailed discussion of the staff's evaluation of this issue appears in the Addendum to the Safety Evaluation Report for NEDC-33083P, Supplement 1, Section 6.0. GEH provided sufficient information on the xenon assumption for startup. Therefore, based on the applicant's responses, RAIs 21.6-82, 4.4-59, and 4.4-60 were resolved.

21.6.3.2.15.6 Decay Heat Modeling

LOCA

During an audit of TRACG as applied to the ESBWR LOCA, the staff reviewed several documents detailing the procedures and calculations performed to determine the shutdown power curve presented in ESBWR DCD, Tier 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. Earlier DCD revisions contained an inconsistent description of the decay heat standard used in the TRACG model, and in RAI 6.3-80, the staff requested clarification. In response to RAI 6.3-80, GEH clarified a typographical error and stated that the decay heat curve was based on the American National Standards Institute/American Nuclear Society (ANSI/ANS) Standard 5.1-1994, "Decay Heat Power in Light Water Reactors," and that there are no differences between the method reviewed in the TRACG audit and the decay heat curve used for ECCS performance analysis in the DCD. Based on the applicant's response, RAI 6.3-80 was resolved.

In RAI 6.3-62, the staff requested additional decay heat modeling details. By letter dated August 17, 2007, the applicant submitted its response to RAI 6.3-62. The response detailed the power used in the LOCA analysis. The ESBWR decay heat was generated based on ANSI/ANS Standard 5.1-1994 with additional terms for a more complete shutdown power assessment. The heat sources in the model include decay heat from fission products, actinides and activation products, as well as fission power from delayed and prompt neutrons immediately after shutdown. The effect of neutron capture in fission products is considered. GEH assumed an end-of-cycle exposure of 32 GWD/ST and an irradiation time of 3.8 years for decay heat calculation. Irradiation time is the most sensitive input in the decay heat model. Increasing irradiation time results in increased contributions from the long-lived actinides, which thus results in higher shutdown powers. The irradiation time of 3.8 years is approximately 8 percent higher than the average batch in the equilibrium cycle. Therefore, the staff considers the assumption used for the decay curve to be conservative.

In addition, GEH provided assumptions related to scram delay times, which included instrument detection of the plant parameters and delay from signal processing. In RAI 6.3-62, the staff also asked GEH to justify the use of the same decay heat curve for both small- and large-break LOCAs. GEH provided a power comparison between the MSIV closure transient at end of cycle and a decay curve used in the LOCA analysis. In the MSIV closure transient, a power increase is experienced at the beginning due to negative void feedback compared to power response in a small-break LOCA. The comparison showed that the decay heat curve bounds the MSIV transient power curve, which implies that the decay curve will bound the small-break LOCA as well and that the decay curve used is conservative. However, from the RAI discussion, the staff noted that the assumptions for the scram signal delay time are different in the MSIV closure transient than in the LOCA event. By estimating additional energy for the small-break LOCA and considering the scram delay time, the additional energy could boil off an additional volume of water in the vessel. The staff estimated this additional volume of water by subtracting this amount of water from GEH's minimum level prediction and then estimated a new minimum water level. The estimated minimum water level is still above the top of the active core. Considering that the decay power calculation includes other conservative assumptions, the staff accepts GEH's approach of using a single decay curve for all LOCA analyses. Based on the applicant's response, RAI 6.3-62 was resolved.

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," issued April 2006, 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 heat structures in TRACG account for this.

The staff finds that GEH has adequately accounted for all major contributors to the decay heat model and that the decay heat 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 of this small a contribution has little or no effect on the overall transient response.

21.6.3.2.16 Numerics

TRACG numerics are a significant improvement over those of 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.

The adequacy of the TRACG field equations and the solution methods have been assessed in NEDE-33083P-A. However, the staff recently identified an error in the TRACE code momentum equation, which overpredicts pressure drops. The staff further concluded that the error was in the original TRAC-P and TRAC-B, which were the basis of the TRACE and TRACG codes, respectively.

In RAI 21.6-123, the staff requested that GEH determine the magnitude of the error in the TRACG momentum conservation equations for the flow from the downcomer through lower plenum to core inlet for the ESBWR configuration. In addition, the staff asked GEH to provide an assessment of the impact of the error in momentum formulation for the analysis supporting the ESBWR DCD and topical reports. GEH submitted the response to RAI 21.6-123 on January 8, 2009, in the letter MFN 09-002. In that response, GEH provided analysis of the ESBWR inlet plenum configuration and the effect that the identified three-dimensional momentum equation error may have on ESBWR safety analysis. GEH concluded the following:

> TRACG models have a deficiency. When calculating 3D flow (e.g. in the lower plenum region), TRACG over-estimates the pressure drop when there are 90 degree flow turns. By analysis, GEH has concluded that the magnitude of this over-prediction is proportional to the density times the velocity squared; thus at low velocities, the over-prediction is small. GEH has quantified an upper estimate of this error in the lower plenum (single phase) pressure drop for ESBWR conditions.

The staff performed a series of simple LAPUR steady-state calculations to quantify the effect of this additional and unphysical TRACG pressure drop on the ESBWR core flow. The staff estimates that the ESBWR flow should be approximately 1.5 percent higher than the flow calculated by TRACG (if the additional unphysical pressure drop is removed). Therefore, for most events, the error adds conservatism.

In response to RAI 21.6-123 (MFN 09-002), GEH evaluated the impact on ESBWR accidents and concluded that there is no significant impact in the safety analysis due to lower flow in the ESBWR as detailed below:

- AOOs: GEH performed a core flow reduction sensitivity analysis using TRACG and observed no significant effect on any AOO. The flow reduction used by GEH bounds the staff's estimation of 1.5-percent flow reduction related to the three-dimensional momentum equation error. The staff accepts that the momentum equation error has a negligible impact on AOOs.

- ATWS: GEH quoted an analysis stating that the core flow reduction due to channel spacer loss increases with a much larger scale compared to the flow reduction due to the momentum error and has only a minimal effect on ATWS performance. Therefore, the core flow reduction caused by the lower plenum DP overprediction is expected to have a minimal effect on ATWS performance. The primary reason for this result is that the ESBWR has significant margin for all three ATWS acceptance criteria: peak cladding temperature (PCT), vessel overpressure, and containment integrity. Since significant margin is available, the staff concurs that flow reduction attributable to the momentum error has no significant impact.

- LOCA: The lower plenum DP error will have no significant impact on the RPV level and the long-term containment pressure in the LOCA analyses. For the LOCA events, GEH argued that the power output drops very quickly after the reactor scram. The natural circulation flow through the downcomer and core reduces correspondingly.

  In the TRACE momentum equation qualification study, the NRC staff corrected the momentum equation error and confirmed that the impact on the RPV level is negligible in LOCAs. Therefore, the staff agrees that the impact on LOCA analysis is small.

- Stability: GEH argued that the RPV lower plenum pressure drop overprediction would lower the core flow, which would be a conservative assumption for stability because lower core flow causes a higher power/flow ratio and leads to a higher decay ratio. This error may have a significant nonconservative effect on the stability of the core-wide stability mode. The following sections discuss the resolution of this concern.

The staff performed sample stability calculations with the LAPUR6 code to evaluate the impact of the TRACG three-dimensional momentum equation error. The issue is that the additional pressure drop introduced by TRACG in the lower plenum is a single-phase pressure drop that tends to stabilize the reactor. This TRACG error is equivalent to unphysical increase in pressure drop to the channel inlet. It is well known that reducing the size (i.e., increasing the friction) of the channel inlet orifice is one of the best methods to stabilize the core.

The staff notes that the unphysical TRACG single-phase pressure drop applies mostly to the outlet of the downcomer, where the flow velocity is higher. Since the out-of-phase (regional) stability mode involves core flow redistribution in the lower plenum, the downcomer flow does not oscillate for regional oscillations; thus, this TRACG error will not significantly affect the regional decay ratio. Some effect will be present as TRACG overestimates the pressure drop caused by the radial component of the lower plenum flow, but this component contribution is small when compared to the vertical flow component contribution in the downcomer. The staff was concerned that the TRACG three-dimensional momentum equation error may significantly impact the stability of the corewide stability mode in the nonconservative direction. GEH's evaluation did not include this effect, so the staff then requested clarification in RAI 21.6-123, Supplement 1.

In the response to RAI 21.6-123, Supplement 1, GEH reiterated the root cause of the three-dimensional momentum equation formulation error. This error is a deficiency of the formulation, which does not take into account the momentum changes by vessel boundaries. The applicant performed a series of stability calculations as a function of lower plenum pressure drop. The applicant also simulated the pressure drop increases by adding an artificial friction factor to the lower plenum components. The resultant calculations indicate that increasing the lower plenum pressure drop has two competing effects: (1) the core flow is decreased, which has a conservative (destabilizing) effect, and (2) the channel inlet friction is increased, which has a nonconservative (stabilizing) effect. The two effects counteract one another, and the decay ratio does not vary significantly when the lower plenum friction is varied from a low to a higher value above the estimated pressure drop error.

The staff reviewed a series of scoping TRACE calculations, where the lower plenum friction was increased to result in an additional pressure drop. The TRACE calculations indicate the same trend as the applicant's TRACG calculations, and no significant effect is seen in the calculated decay ratio. Based on these analyses, the staff concludes that the decreased core flow and the increased channel inlet friction have competing effects and that the resulting net effect on the decay ratio calculated by TRACG is not affected significantly by the three-dimensional momentum equation formulation error. Thus, based on the applicant's response, RAI 21.6-123 was resolved.

21.6.3.2.17 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 simulation is missing. The heat exchanger model contains some simplifying approximations that may not be appropriate for simulating the IC or the condenser in balance of plant. 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.

21.6.3.2.17.1 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.

21.6.3.2.17.1.1 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 shows core uncovery, the core does not heat up, and GEH did not calculate a PCT increase, 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-32177, Revision 3, to be 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 had addressed these characteristics and that the procedure 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.

The SER for the application of General Electric's LTR, "TRACG Application for ESBWR Stability Analysis," NEDE-33083P, Supplement 1, discusses the staff's review of the channel grouping and nodalization for ESBWR stability analyses in detail.

AOO/IE and ATWS

For the ESBWR AOO/IE and ATWS events, GEH used the same axial nodalization for the channel component as that used for the ESBWR stability analysis (NEDC-33083P, Supplement 1). GEH combines the channels based on similar hydrodynamic, as well as neutron kinetic, 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," issued 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 IE. 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, Supplements 2 and 3, adequately represents the ESBWR core and is acceptable.

The staff requested information supplemental to RAI 21.6-65 on the channel grouping used for ESBWR AOO/IE evaluations. The staff asked whether GEH used only the hot channels for calculating the ΔCPR/ICPR because the staff was concerned that, although this is most conservative in many cases, for cold-water injection events, the ΔCPR/ICPR may be underestimated because the largest change in CPR would take place in the periphery channels. In MFN 08-340, GEH responded to RAI 21.6-65, Supplement 1, by noting that the hot channel is always selected for the determination of the operating limit MCPR, because the core MCPR always occurs in the hottest channels. The response in MFN 08-340 also shows MCPR calculations for the limiting cold water injection event. The results for the limiting channel in Ring 3 and the hottest bundle in Ring 2 are shown and compared. Although the decrease in ΔCPR/ICPR is greater in the Ring 3 channel, the MCPR in the hottest bundle is still limiting by a significant amount. The staff finds this response, and therefore the channel grouping for AOOs, to be acceptable. Based on the applicant's response, RAI 21.6-65 was resolved.

21.6.3.2.17.1.2 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 the 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 RAIs 4.4-11 and 4.4-12, GEH performed a series of detailed analyses of the effect of the chimney on two instability modes—density wave and loop instability. 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 GEH's response to RAI 4.4-11. 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 GEH's response to RAI 4.4-11 compares the core power response to a flow perturbation. Again, the responses for the coarse and fine nodalizations in the chimney closely agree.

Figure 4.4-11-5 in GEH's response to RAI 4.4-11 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 RAI 4.4-11 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 issued a supplement to RAI 4.4-58, and the applicant submitted its response in MFN 08-542, dated July 2, 2008. In the second part of the response to RAI 4.4-58, Supplement 1, the applicant justified the use of coarse nodalization in the chimney. The applicant's argument is that the chimney does not play an important role in the density-wave instabilities of interest. Loop oscillations (where the chimney plays an important role) are not limiting in the ESBWR and do not pose any significant safety concern. The applicant concluded that the coarse chimney nodalization is adequate for the ESBWR stability analysis.

After reviewing the available data, the staff finds that, for the density-wave oscillations that are likely to be limiting in the ESBWR, the chimney does not appear to play a significant dynamic role; thus, numerical damping in the chimney region is not likely to affect the magnitude of the calculated decay ratio. Therefore, the staff concurs with the applicant's evaluation and accepts that a coarse chimney nodalization is sufficient to model density-wave oscillations. Based on the applicant's response, RAIs 4.4-11, 4.4-12, and 4.4-58 were resolved.

21.6.3.2.17.1.3 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.

21.6.3.2.17.2 Isolation Condenser System

The ICS provides additional liquid inventory to the RPV during a LOCA upon opening the condensate return valves to initiate the system. The ICS also provides the reactor with depressurization for AOO/IE and ATWS pressurization events such as an MSIV closure.

GEH performed a series of steady-state and transient IC tests and compared the data to 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 noncondensable gas buildup and a pool water level transient. The IC testing was performed at the PANTHERS-IC test facility. Section 4.2 of NEDC-32725P, Revision 1, describes the tests and the TRACG comparisons. Section 3.2 of MFN 04-059. The staff reviewed the results of these tests and the modeling of the ICS in the ESBWR TRACG input decks. RAI 21.6-55 asked a series of questions about the applicability of the ICS tests and the corresponding TRACG runs for the ESBWR.

The staff's review of the responses to RAI 21.6-55 and related supplements is discussed in detail in "Safety Evaluation, Application of the TRACG Computer Code to Anticipated Operational Occurrences and Infrequent Events for the ESBWR Design NEDE-33083P, Supplement 3."

One item related to RAI 21.6-55 that is not discussed in the staff SER for NEDE-33083, Supplement 3, is the capability of TRACG to model IC behavior when noncondensable gases are present for LOCA events. (The staff finds that TRACG is capable of modeling the IC behavior during AOO/ATWS and LOCA events for purely steam condensation.)

The staff agrees with GEH that it is not necessary to model noncondensable gases generated by radiolytic decomposition for ESBWR AOO/ATWS calculations because the IC is vented during normal operation and the duration of these transients is not long enough to generate these gases. For the most recent ESBWR LOCA analyses, the heat removal capacity of the ICS is not credited. Therefore, the ESBWR LOCA results are conservative and acceptable. Based on the applicant's response RAI 21.6-55 was resolved.

21.6.3.2.17.3 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. This item is discussed in detail in the staff's Safety Evaluation for the Application of the TRACG Computer Code to Anticipated Transients without SCRAM for the ESBWR Design (NEDE-33083P, Supplement 2). The staff finds the SLCS modeling to be acceptable.

21.6.3.2.17.4 Containment Components

The SER approving NEDC-33083P-A documents the staff's evaluation of TRACG for containment analysis. 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 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 set for 4,500-MW core power. DCD,Tier 2, Revision 4, Tables 6A-1 and 6.2-6a and Appendix 6B, document these design and modeling changes. Section 6.2.1 of this report presents the staff's evaluation of these changes.

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 appears in the Addendum to the Safety Evaluation Report for NEDC-33083P-A, Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design. As discussed in the addendum to the staff SER for NEDC-33083P-A, all of the confirmatory items are closed.

The following sections describe 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 Section 6.2 of this report.

21.6.3.2.17.4.1 Wetwell

The wetwell consists of the 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 PCC vents is also discharged to the SP.

Wetwell Gas Space

The wetwell gas (steam and noncondensables) 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, noncondensable 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 noncondensable gases, temperature distribution, thermal stratification, and pool two-phase level.

21.6.3.2.17.4.2 Passive Containment Cooling Pools

The six PCC pools are located outside (above) the containment. The total PCCS cooling capacity is 66 MW. Each pool 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 documents this review.

21.6.3.2.17.4.3 Passive Containment Cooling Units

The ESBWR has six PCC heat exchanger units. Each comprises 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. One-dimensional components simulating the inlet piping, headers, condenser tubes, condensate discharge lines, and ventlines, represent the PCC units. 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 ventline outlet is higher than the outlet of the upper horizontal drywell/wetwell LOCA vents. Noncondensable gases and uncondensed steam are vented to the SP. 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 documents this review.

21.6.3.2.17.4.4 Horizontal Vent System

The ESBWR has 36 horizontal vents between the drywell and the SP. The 12 vertical flow channels each contain 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 documents the review of the horizontal vents.

21.6.3.2.17.4.5 Gravity-Driven Cooling System Equalizing Lines

As described in ESBWR DCD, Tier 2, Revision 7, Section 6.3.2.7.1, 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. 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 documents the review of the GDCS equalizing lines.

21.6.3.2.17.4.6 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 exceeds that of the drywell by a specified value. The VBs are represented by one-dimensional VALVE components. The VBs are lumped together as one component and 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 documents the review of the vacuum breakers.

21.6.3.2.17.4.7 Reactor Pressure Vessel Level versus Minimum Containment Pressure

During the December 2006 audit, the NRC staff and GEH discussed the use of the TRACG containment model in the RPV-level calculation. The staff was concerned that the containment portion of the input deck was designed with assumptions to give maximum containment pressure, which may be nonconservative for RPV-level calculations, as historical operating plant analyses have been performed with minimum containment pressure. At that time, no conclusion was reached as to the impact that the minimum containment pressure would have on the ESBWR RPV-level calculation. The staff addressed this issue in RAI 6.2-144. In response, GEH evaluated the impact of containment back pressure on the ECCS performance and updated the results 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. Based on the applicant's response, RAI 6.2-144 was resolved.

21.6.3.2.17.4.8 TRACG Modeling of Steam Source

During a December 2006 audit, GEH provided the results of sensitivity studies in which it examined the effects of placing the steam source at 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 these results are inconsistent with the current GEH practice of placing the steam source below the RPV. GEH addressed this particular issue in response to RAI 6.2-53, Supplement 1. In its response, GEH showed sensitivity analyses results to confirm that the current bounding MSLB case where the break occurs at Level 34 is limiting. For calculations in ESBWR DCD, Tier 2, Revision 2, GEH has used an 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 ESBWR DCD, Tier 2, Revision 3. Based on the applicant's response, RAI 6.2-53, Supplement 1, including all supplemental questions for that RAI, was resolved.

21.6.3.3 Accident Scenario Identification Process

The behavior of a nuclear power plant undergoing an accident or transient is not influenced equally by all phenomena that occur during the event. Those phenomena that are important for each event and the various phases within an event must be determined. 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 Loss-of-Coolant Accident

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 the ESBWR LOCA acceptable because the assessment of the code included all of the high- and medium-ranked phenomena. NEDC-33083P-A includes more details of the staff's review of the PIRT for LOCA.

21.6.3.3.1.1 Loss-of-Coolant Accident Long-Term Core Cooling

GEH letter MFN 05-105, dated October 6, 2005, provides details on long-term core cooling. The letter discusses 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 to the Safety Evaluation NEDC-33083-P-A, "TRACG Application for ESBWR," provides more details of the staff's review of the long-term core cooling PIRT. The staff finds the GEH PIRT acceptable for demonstrating long-term core cooling of all LOCA cases, and all important components are modeled.

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. The 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 Anticipated Operational Occurrences/Infrequent Events

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 review of the AOO PIRT is in the SER for "Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design."

In RAI 21.6-61, the staff questioned why mixing in the lower plenum for cold-water injection events was not ranked as high. The staff's concern was that as cold water enters the core, it might not mix well in the lower plenum, and there may be areas of concentrated cold water causing a pronounced effect on ΔCPR for bundles in this location. The staff was concerned that the impact on ΔCPR would be greater than that calculated by TRACG because of the coarse noding of the lower plenum in the ESBWR TRACG model at the time that RAI 21.6-61 was transmitted. A PIRT ranking of high would ensure that the uncertainties associated with lower plenum mixing are included in the calculation of ΔCPR.

The GEH response to RAI 21.6-61 included 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 the ultimate change in ΔCPR, given that the different resistances in the lower plenum did not have a substantial effect on ΔCPR for Ring 1 (the most central) and Ring 2 (next to the most central). GEH did not display the results of the ΔCPR changes for Ring 3, which is the outermost ring and the one of greatest concern.

Subsequent to the GEH response to RAI 21.6-61, the ESBWR DCD, as well as NEDE-33083, Supplement 3, shows that the PIRT ranking for lower plenum mixing is high, and the TRACG AOO methodology for the ESBWR accounts for the corresponding uncertainty. In addition, a computational fluid dynamics study performed by the staff for the limiting cold-water injection event cold-water transient shows that the thermal mixing at the side entry orifices (which is representative of the mixing that occurs in the downcomer and lower plenum) is consistent with the results from the TRACG model. Therefore, RAI 21.6-61 was closed.

21.6.3.3.4 Anticipated Transient without Scram

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 on 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. The safety evaluation for the "Application of the TRACG Computer Code to Anticipated Transients without Scram for the ESBWR Design NEDE-33083P, Supplement 2," discusses the staff's review in detail.

21.6.3.4 <u>Code Assessment</u>

The staff reviewed the following assessment cases that were analyzed 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 low-pressure coolant injection 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. The staff has reviewed other tests as part of the TRACG review, and these reviews 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.

The staff requested in RAI 21.6-75 that GEH provide the TRACG04 version used to perform analysis of all ESBWR LOCA events in the design certification documentation. In response, GEH submitted Revision 3 of the ESBWR qualification report (NEDC-32177). The staff reviewed the GEH qualification of its void fraction data provided in this report to ensure that the modifications to the entrainment fraction and its subsequent use in the interfacial shear model compare well with the data. The void fraction assessment results from NEDE-32177P, Revision 3, are very close to the results from NEDE-32177P, Revision 2, which was assessed as satisfactory during the ESBWR preapplication phase of the ESBWR design certification review. This ensures that the conclusion from the preapplication TRACG review is still valid. In addition, NEDE-32177P, Revision 3, adds assessment cases which include Toshiba Low-Pressure Void Fraction Tests, Ontario Hydro Void Fraction Tests, and CISE Density Measurement Tests. The Toshiba tests were added to extend the qualification basis to lower pressures at 0.5 and 1.00 MPa. The Ontario Hydro facility provides void fraction data from a large-scale pumped flow facility. The CISE facility in Italy provided data related to the void and quality relationship. The TRACG assessment showed good agreement with the data from those tests. The assessment from NEDE-32177P, Revision 3, reinforced the conclusion from the approved GEH LTR NEDC-33083P-A that the interfacial shear model is acceptable. Based on the applicant's response, RAI 21.6-75 was resolved.

21.6.3.5 Uncertainty Analysis

21.6.3.5.1 Loss-of-Coolant Accident

In the letter MFN 05-096, dated September 9, 2005, GEH stated that since there is no core heatup, an uncertainty analysis of PCT would not provide useful results. GEH further stated 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. GEH responded to RAI 6.3-81 by stating that, because there is no core uncovery and no core heatup for the ESBWR LOCAs, a statistical analysis of the PCT serves no useful purpose. The best estimate PCT and the 95/95 PCT would both be close to the saturation temperature corresponding to the peak steam dome pressure reached in the accidents. For the case of ESBWR LOCAs, there is a margin of over 1,600 degrees F to the limit of 2,200 degrees F (acceptance criteria set forth in paragraph (b) of 10 CFR 50.46). GEH further commented that the static head inside the chimney (in meters of water) is selected as the figure of merit for comparison and for evaluating the impact of uncertainties in model parameters and plant parameters. This collapsed level is defined as the equivalent height of water corresponding to the static head of the two-phase mixture above the top of the core. NEDC-33083P-A, Sections 2.4 and 2.5.3, identify the TRACG model parameter uncertainties and plant parameter uncertainties. Sensitivity studies were performed by varying each of these parameters from the lower bound to the upper bound value. The impact on the chimney static head is between -0.3 meters to +0.2 meters (NEDC-33083P-A, Section 2.4.4.2), which is less than the minimum static head in the chimney from the parametric studies. Therefore, GEH proposed that a simple calculation be made, setting the most significant parameters at the 2-sigma values to obtain a bounding estimate of the minimum level.

The staff concurs that the ESBWR LOCA results demonstrate a high level of probability that there is no core uncovery or heatup and that the PCT would be close to the saturation temperature corresponding to the peak steam dome pressure reached in the accidents and therefore would not exceed the acceptance criteria of 10 CFR 50.46. The staff concludes that GEH's LOCA results comply with the requirement of 10 CFR 50.46. Based on the applicant's response, RAI 6.3-81 was resolved.

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 1-sigma uncertainty values of the high- and medium-ranked parameters, which are discussed in the SER for NEDE-33083P, Supplement 1. The staff reviewed the uncertainties in the physics parameters as part of design certification; the addendum to the SER for NEDE-33083P, Supplement 1, discusses these uncertainties.

21.6.3.5.3 Anticipated Operational Occurrence/Infrequent Event

For the TRACG application for ESBWR AOO analysis described in NEDC-33083P, Supplement 3, GEH used the same method for determining combined bias and uncertainty as is used in the GEH application of TRACG to the BWR/2–6 AOO analysis (NEDE-32906P-A, Revision 2). The staff previously reviewed this method in detail and accepted it for the application of TRACG to BWR/2–6 AOO analyses. The associated SER documents the staff's review for BWR/2–6 AOO analyses.

For the ESBWR AOO analyses in NEDC-33083P, Supplement 3, GEH either bounds or accounts for the uncertainty of all high- and medium-ranked PIRT parameters. For parameters with uncertainty accounted for, GEH chose the ND-OSUTL method if the output distribution is normal and used the order statistics method if the output distribution is not normal. The SER for NEDC-33083P, Supplement 3, presents the staff's review of TRACG for ESBWR AOO analyses. The staff finds the use of this methodology acceptable for determining the combined uncertainty for TRACG modeling of the ESBWR AOO events.

21.6.3.5.4 Anticipated Transient without Scram

The method that GEH uses to combine uncertainties for the ESBWR ATWS events is different than the method used for the analysis of other events. For the ESBWR ATWS analyses in NEDE-33083P, Supplement 2, GEH either bounds or accounts for biases and uncertainties for all high-ranked PIRT parameters. For the parameters with uncertainty accounted for, the propagation of errors method is used to combine uncertainties.

The staff has reviewed this method and finds it acceptable for determining the combined uncertainty for TRACG modeling of ESBWR ATWS events. The safety evaluation for NEDE-33083P, Supplement 2, discusses the staff's review in greater detail.

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 confirmatory calculations for the LOCA water-level events and documented the results in Section 6.3 of this SER. The confirmatory calculations show that the key parameter responses during LOCA by GEH are reasonable and the core remains covered with water in all LOCA analyses.

21.6.5 Conclusions

The staff reviewed the TRACG code based on the review guidelines of SRP Section 15.0.2, issued December 2005. The evaluation covered required documentation: (1) the evaluation models, (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 staff concludes that TRACG, including the application methodology, is an acceptable evaluation model for ESBWR LOCAs, AOOs/IEs, ATWS, and stability. These methods must follow the application procedures outlined in the associated topical reports and adhere to the NRC conditions and limitations specified in the staff SERs.

21.7 Quality Assurance Inspection

The staff relied on four principal test programs (PANDA, PANTHERS/PCCS, PANTHERS/ICS, and GIRAFFE) to demonstrate that analytical methods and computer codes described in this chapter were adequately and appropriately applied in ESBWR accident analyses. 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 timeframe, 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 conducted these inspections to determine if GEH and its contractors/partners had fulfilled GEH's commitment to its DCD, Chapter 17 QA requirements. ESBWR DCD Tier 2, Section 1.5, and Section 21.3 of this report describe these test programs in detail. 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 described the corrective actions taken in its responses to these NRC inspection findings. The staff accepted these responses and corrective actions as responsive to the NRC inspection findings.

As described in Section 17.0 of this report, the applicant has continuously maintained a QA program that meets the requirements of Appendix B to 10 CFR Part 50 and spans 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 that meet 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 Unresolved Item (URI) 05200010-2005-201-02 below.)

As part of its review of DCD Chapter 17, the staff inspected the implementation of the GEH QA program for ESBWR activities. The staff performed these inspections in November 2005, April 2006, and December 2006. The following describe the findings:

- David B. Matthews (NRC) to David H. Hinds (GENE), "NRC Inspection Report 05200010/2005-201 and Notice of Nonconformance," January 11, 2006

- David B. Matthews (NRC) to David H. Hinds (GENE), "NRC Inspection Report 05200010/2006-201 and Notice of Nonconformance," June 14, 2006

- David B. Matthews (NRC) to David H. Hinds (GENE), "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

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 design certification testing programs performed for the SBWR that are also being used to support design certification of the ESBWR.

GEH responded in the following letters:

- David H. Hinds (GENE) to NRC Document Control Desk, "Reply to Notice of Nonconformance NRC Inspection Report 05200010/2005-201, dated January 11, 2006," February 9, 2006.

- MFN 06-053, Enclosure 2, "Quality Oversight for the SBWR Test Program," to Supplement 1, dated October 27, 2006, 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 (GENE) to NRC Document Control Desk, "NRC Inspection 05200010/2005-201 Unresolved Items," October 27, 2006. This response described the quality oversight activities that had been performed and identified the NRC and GEH documentation supporting the SBWR test program inspections.

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 concludes that the QA programs governing GEH's SBWR/ESBWR design certification test programs satisfy the requirements of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," and the pertinent provisions of Appendix B to 10 CFR Part 50.

21.8 References

American National Standards Institute/American Nuclear Society (ANSI/ANS)

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GE-Hitachi Nuclear Energy Americas LLC (GEH)

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— — — — —, GE Nuclear Energy, NEDC-33081P, Class III, Revision 1, “ESBWR Test Report,” May 2005 (ADAMS Accession Nos. ML051880441, ML051880444).

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