HITACHI

GE Hitachi Nuclear Energy

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Licensing Topical Report

TRACG APPLICATION FOR ESBWR

Public Information Notice

This is a public version of NEDC-33083P-A Revision 1 from which the proprietary information has been removed. Portions of the document that have been removed are indicated by white space within double square brackets, as shown here [[ ]].

SUMMARY OF CHANGES
NEDO-33083-A Revision 1

Location	Comment
Cover sheet	Revised cover sheet
SE for Rev 0	The SE including RAI responses associated with NEDO-33083-A was removed from the document
All	"-A" is added to the document number for this revision denoting NRC acceptance of this revision for ESBWR design certification.
Attachment 1	Added the NRC letter describing the final acceptance of Revision 1 of this Licensing Topical Report. Whereas revision 0 of this document was reviewed by the NRC, revision 1 was necessary because of the prior "-A" acceptance status of the document. The NRC letter as well as the Enclosure 1 of the letter, which contains the Final Safety Evaluation Addendum to the SE for this Licensing Topical Report, has been added to the end of this document as Attachment 1.

Table of Contents

Table of Contents (cont'd)

Table of Contents (cont'd)

Table of Contents (cont'd)

Table of Contents (cont'd)

List of Tables

List of Tables (cont'd)

List of Figures

List of Figures (cont'd)

List of Figures (cont'd)

Page

1. INTRODUCTION

1.1 Background

TRACG is a General Electric (GE) proprietary version of the Transient Reactor Analysis Code (TRAC). TRACG uses advanced realistic one-dimensional and three-dimensional methods to model the phenomena that are important in evaluating the operation of BWRs. Realistic analyses performed with TRACG have been used previously to support licensing applications in different areas, including transients otherwise known as an Anticipated Operational Occurences (AOO), and pipe breaks referred to by the acronym ECCS/LOCA (Emergency Core Cooling Systems/Loss of Coolant Accident). Recently, the application of TRACG for Anticipated Operational Occurrences (AOOs) for operating BWRs has been approved by the NRC [3].

TRAC was originally developed for pressurized water reactor (PWR) analysis by Los Alamos National Laboratory, the first PWR version of TRAC being TRAC-P1A [4]. The development of the BWR version of TRAC started in 1979 in close cooperation between GE and Idaho National Engineering Laboratory. The objective of this cooperation was the development of a version of TRAC capable of simulating BWR LOCAs. The main tasks consisted of improving the basic models in TRAC for BWR applications and in developing models for specific BWR phenomena and components. This work culminated in the middle 1980's with the development of TRACB04 at GE [5],[6],[7],[8],[9],[10],[11] and TRACG-BD1/MOD1 at INEL [12]. Due to the joint development, these versions were very similar. In the earlier stages, General Electric (GE), the United States Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute (EPRI) jointly funded the development of the code. A detailed description of these earlier versions of TRAC for BWRs is contained in References 12 through 14.

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1.2 Summary

The TRACG computer code is used to perform licensing analysis of the ESBWR. This report presents the methodology for application of TRACG to the ESBWR. TRACG is specifically used for the following four categories of analyses:

1. ECCS/LOCA
2. Containment/LOCA
3. Anticipated transients with scram (AOO)
4. Anticipated transients without scram (ATWS)

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1.2.1 ECCS/LOCA Application

LOCA events (Section 2) are analyzed to establish the reactor system response, including the calculation of the chimney level and Peak Cladding Temperature (PCT). Because there is no core uncovery for any break size or location, local cladding oxidation and core-wide cladding oxidation do not need to be evaluated. This application specifically addresses TRACG capabilities to ensure that TRACG is a qualified model for evaluating margins to the acceptance criteria for ECCS performance stated in 10CFR50.46. The application report defines the application process and demonstrates that TRACG analyses can be used for ECCS/LOCA licensing calculations. The application process includes the quantification of uncertainties that are applied to the realistic nominal results of TRACG analyses, resulting in a “licensing calculation”.

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1.2.2 Containment/LOCA Application

TRACG is utilized for the calculation of the containment pressure and temperature transient (Section 3). The application methodology will be used to demonstrate that the containment and its associated systems can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

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1.2.3 AOO Application

This document describes the application methodology for AOOs (Section 4) that is in compliance with licensing limits. AOO events are analyzed to establish the reactor system response, including the calculation of the Operating Limit Minimum Critical Power Ratio (OLMCPR). TRACG capabilities are addressed to ensure that TRACG is a qualified model for the evaluation of margins to acceptable fuel design limits and reactor coolant pressure boundary design conditions. This application report extends the approved TRACG application methodology for AOO analysis to the ESBWR. Uncertainties are quantified and will be applied to the realistic nominal results of TRACG analyses. The licensing criteria to be satisfied is that less than 0.1% of the fuel rods are expected to experience a boiling transition for the most severe AOO.

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Some of these uncertainties are fuel type dependent. Therefore, periodic changes in the statistical analysis will be required as core design changes. The statistical analysis process is defined in this report and criteria to be used to change this analysis are provided.

The overall analysis approach followed is consistent with the Code Scaling Applicability and Uncertainty (CSAU) analysis methodology [28]. Conformance with CSAU methodology is demonstrated in Section 4.1.3.1.

1.3 Scope of Review

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The Licensing Topical Reports NEDE-32176P, TRACG Model Description [1]; NEDE-32177, TRACG Qualification [2]; NEDC-32725, TRACG Qualification for SBWR Volumes 1 and 2, [24] and NEDC-33080, TRACG Qualification for ESBWR [25] are incorporated by reference as part of the review scope.

2. ECCS/LOCA ANALYSIS

2.1 Licensing Requirements and Scope of Application

2.1.1 General Requirements

The *General Design Criteria (GDC) for Nuclear Power Plants* are stipulated in Appendix A to Part 50 of 10CFR. The applicable GDC is GDC 35, which requires each BWR to be equipped with an emergency core cooling system (ECCS) that refills the vessel in a timely manner to satisfy the requirements of the regulations for ECCS performance given in 10 CFR Part 50, §50.46 and Appendix K to 10CFR50 [17]. GDC 35 also requires redundant ECCS components to be provided to adequately cool the core during a LOCA. 10CFR100 [18] specifies mitigation of radiological consequences of an accident. Guidance is also provided in 10CFR 50.34 *(Contents of Applications; Technical Information)*.

2.1.2 Specific 10CFR50.46 Licensing Acceptance Criteria for ECCS Performance

The specific 10CFR50.46 licensing acceptance criteria for ECCS performance are as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
2. The calculated total local oxidation of the cladding shall nowhere exceed 17% of the total cladding thickness before oxidation.
3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 1% of the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by the long-lived radioactivity.

2.1.3 Analysis Requirements

The calculational framework used for evaluating the ECCS in terms of core behavior is called an evaluation model. It includes one or more computer programs, the mathematical models used, the assumptions and correlations included in the program, the procedure for selecting and treating the program input and output information, the specification of those portions of the analysis not included in computer programs, the values of parameters, and all other information necessary to specify the calculation procedure. The evaluation model must comply with the acceptance criteria for ECCS given in 10CFR50.46 and Appendix K to 10CFR50. The evaluation model must have been previously documented and reviewed and approved by the NRC staff.

On September 16, 1988, the NRC staff amended the requirements of §50.46 and Appendix K so that these regulations reflect the improved understanding of ECCS performance obtained through the extensive research performed since the promulgation of the original requirements in January 1974. Paragraph 50.46 (a)(1) now permits the use of a realistic evaluation model. It also requires that the uncertainty in the realistic evaluation model be quantified and considered with the applicable limits in Paragraph 50.46 (b) listed above, so that there is a high probability that the criteria will not be exceeded. Regulatory Guide 1.157 [16] describes models, correlations, data, model evaluation procedures, and methods that are acceptable to the NRC staff for a realistic or best-estimate calculation of ECCS performance during a LOCA and for estimating the uncertainty in that calculation. Both the NRC and ACRS have stated that the CSAU methodology [15] is in full compliance with Regulatory Guide 1.157. Compliance of the GE methodology for ECCS/LOCA analysis with Regulatory Guide 1.157 is demonstrated in Section 2.1.5.1. Conformance with the CSAU process is shown in Section 2.1.5.2.

2.1.4 Standard Review Plan (SRP) Guidelines (NUREG 800)

The NRC guidelines for review of ECCS/LOCA safety analysis are identified in Section 15.6.5 of the SRP [19], *Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary*. A draft Section 15.0.1, *Review of Analytical Computer Codes,* is currently undergoing NRC review.

2.1.5 Proposed Application Methodology

TRACG is a complete transient thermal-hydraulic model, and it will be used to calculate the entire LOCA transient for both the vessel and containment.

TRACG calculates the PCT, local oxidation and core-wide oxidation. Thus, conformance with Criteria 1 through 3 of 10CFR50.46 is demonstrated by the TRACG analysis results. As discussed in Reference 88, conformance with Criterion 4 (coolable geometry) is demonstrated by conformance to Criteria 1 and 2. The bases and demonstration of compliance with Criterion 5 (long term cooling) are documented in Reference 88, and are usually not affected by the TRACG ECCS/LOCA analysis.

2.1.5.1 Conformance with Regulatory Guide 1.157

The proposed application methodology using TRACG for ESBWR ECCS/LOCA analyses complies with all the requirements of Regulatory Guide 1.157, “Best-Estimate Calculations of Emergency Cooling System Performance” [16]. This section shows how these requirements are addressed on a point-by-point basis.

The regulatory guide describes models, correlations, data, model evaluation procedures, and methods that are acceptable to the NRC staff for meeting the requirements for a realistic or best-estimate calculation of ECCS performance during a LOCA and for estimating the uncertainty in that calculation. It also provides a description of the acceptable features of best-estimate computer codes and acceptable methods for determining the uncertainty in the calculations. The guide lists TRAC-BWR as an acceptable code for best-estimate calculations of ECCS performance. Both the NRC and ACRS have stated that the CSAU process [15] is in full compliance with the Regulatory Guide and is acceptable under the provisions of Paragraph 50.46(a)(1) for use of a realistic evaluation model. The GE methodology follows the CSAU steps (see next section). Thus, the GE methodology should be acceptable with respect to the requirements of Regulatory Guide 1.157. Nevertheless, in this section the features of the GE methodology are compared with the required features in the regulatory positions in Regulatory Guide 1.157.

Regulatory Position 1: Best-Estimate Calculations

Staff Position	GE Process	Evaluation
Licensees may use TRAC-PWR, TRAC-BWR, RELAP5, COBRA and FRAP codes	TRACG, a derivative of TRAC-BWR, is used.	TRACG shares the same structure and field equations as TRAC-BF1. The bulk of the constitutive relations are the same [1]. Differences are listed in Appendix A of Reference 1. TRACG is in the family of acceptable codes.
Licensee must demonstrate that the code and models used are acceptable and applicable to the specific facility over the intended operating range.	Description of models [1] and qualification [2],[24],[25] demonstrate applicability. Range of test data and qualification requirements are specified in these documents.	Range of models and correlations reviewed by NRC in TRACG Model Report [1]. Model acceptability demonstrated through qualification against test data and reviewed by NRC as part of TRACG Qualification [2].
Licensee must quantify uncertainty in the specific application.	Uncertainty is quantified in the application report for ECCS/LOCA application.	Uncertainty obtained from integral comparisons and bounded by combination of individual uncertainties. Meets CSAU and Reg. Guide requirements.
The model should be compared with applicable experimental data and should predict the mean of the data.	TRACG has been compared against a wide range of applicable data and generally predicts mean of data [2],[24],[25].	TRACG is intended to predict mean of data. Bias and uncertainty in predictions are quantified in Qualification Reports.
Effects of all important variables should be considered.	Capability to treat important phenomena is shown in PIRT Section 2.2 of this report.	TRACG considers all important LOCA parameters.

Staff Position	GE Process	Evaluation
Best-estimate code should be compared with applicable experimental data (e.g., separate effects tests and integral simulations of LOCAs) to determine overall uncertainty and bias	Comparisons made in Model Report [1] and Qualification Reports [2],[24],[25] for separate effects and integral tests.	Requirements satisfied.

Regulatory Position 2: Considerations for Thermal-Hydraulic Best-Estimate Codes

2.1.1 Numerical Methods

Staff Position	GE Process	Evaluation
Sensitivity studies and evaluations of the uncertainty introduced by noding should be performed.	ESBWR nodalization is justified through qualification studies and sensitivity studies in the SBWR Qualification Report [24].	Reg. Guide requirements satisfied.
Effect of time step size should be investigated.	Time step is determined internally by TRACG (Section 8.2.4 of Reference 1). Maximum time step has been varied in calculations to show insensitivity [2].	Insensitivity to time step size demonstrated in the range of time steps sizes used for the calculations.

2.1.2 Computational Models

Staff Position	GE Process	Evaluation
Separate flow fields for different fluid phases and calculation of nonequilibrium between phases may be required.	TRACG has separate field equations for the vapor and liquid phases and calculates individual phasic velocities and temperatures [1, Section 3.1.2].	The adequacy of the TRACG field equations and constitutive relations has been validated by extensive comparisons against separate effects data for void fraction and heat transfer [2, Sections 3.1 and 3.2].

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Regulatory Position 3: Best-Estimate Code Features

3.1 Initial and Boundary Conditions and Equipment Availability

Staff Position	GE Process	Evaluation
Most limiting initial conditions expected over the life of the plant should be used.	Most limiting operating conditions (power/flow, pressure, exposure, etc.) have been determined.	Limiting operating conditions are used in analysis.
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The calculations should be performed over the spectrum of possible break sizes up to a full double-ended break of the largest pipe. Effects of longitudinal splits with the split area equal to twice the cross-sectional area of the pipe should be included.	The full spectrum of breaks is analyzed. The split break evaluation has no specific consideration of break geometry; the conditions upstream of the break are determined by flow from both sides of the break location.	The break spectrum is analyzed to identify the case leading to the minimum chimney static head (no core heatup).
Other boundary and initial conditions (equipment availability, control systems and operator actions) should be based on plant technical specification limits.	Trips such as scram, MSIV closure, ADS opening, etc., are assumed to occur based on technical specification limits. Instrument setpoints and equipment performance are set to their analytical limits. The LOCA analysis takes no credit for non-safety systems to mitigate the accident. When the expected operation of a non-safety system can cause the results to be more severe (e.g., bypass valve pressure regulation), it is considered.	Analytical values corresponding to the technical specification limits are used, accounting for uncertainties. No credit is taken for non-safety systems or for mitigating operator actions.

Staff Position	GE Process	Evaluation
Single failure and loss of onsite and offsite power should be considered.	Loss of preferred power is assumed. Sensitivity to all single failures is considered.	Process conforms to Reg. Guide and Appendix A of 10 CFR 50.

3.2 Sources of Heat During a LOCA

3.2.1 Initial Stored Energy of the Fuel

Staff Position	GE Process	Evaluation
The steady-state temperature distribution and stored energy in the fuel should be calculated on a best-estimate basis.	Because the stored energy is dependent on the plant operating history at the time of LOCA, a design basis operating trajectory is used to calculate this parameter.	Reasonable approach, considering operating states.
An acceptable model should recognize the effects of fuel burnup, fuel pellet cracking and relocation, cladding creep, and gas mixture conductivity.	The GESTR **[27]** model includes all of these effects. The TRACG dynamic gap conductance model (Section 7.5.2 of Reference 1) is initialized by GESTR.	GESTR has been separately reviewed and accepted for use by the NRC staff **[27]**.
The model must be checked against several sets of relevant data.	The GESTR model has been extensively compared with irradiated BWR fuel data **[27]**.	GESTR has been separately reviewed and accepted for use by the NRC staff **[27]**.

3.2.2 Fission Heat, 3.2.3 Decay of Actinides, 3.2.4 Fission Product Decay Heat

Staff Position	GE Process	Evaluation
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The heat from radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, should be calculated in accordance with fuel cycle history.	Heat from radioactive decay of actinides, including neptunium and plutonium, as well as isotopes of uranium, is included in the calculation.	The model used is in compliance with Reg. Guide requirements.
The heat generation from radioactive decay of fission products should be calculated in accordance with the 1979 ANS standard.	The heat generation from radioactive decay of fission products is calculated in accordance with the ANS standard. A generic curve is calculated to characterize the core average response for reference values of fuel exposure, depletion power density, irradiation time, fuel enrichment, and void fraction. Uncertainties due to variations in the operational parameters listed above, as well as due to measurements, are considered.	Calculations are made in accordance with the 1979 ANS Standard. The average core decay heat history is slightly conservative for most operating conditions. Sensitivities to variations in voids, enrichment and operating history are shown in Appendix B of Reference 21.

3.2.5 Metal-Water Reaction Rate

Staff Position	GE Process	Evaluation
The metal-water reaction rate should be calculated with a best-estimate model. For rods calculated to rupture, oxidation of the inside of the cladding should be calculated.	The Cathcart correlation (Equation 6.6-136 of Reference 1) is used at all temperatures. The model is also used on the inside surface of the cladding if the fuel rod perforates.	Acceptable model is used. Metal-water reaction is of no importance for ESBWR, as PCTs are low (no cladding heatup). Metal-water reaction is negligible below 1700°F.
Below 1900°F, model should be checked against appropriate data. It should recognize the effects of steam pressure, pre-oxidation of cladding, deformation during oxidation and internal oxidation from both steam and UO_2 fuel.	The Cathcart correlation is used. This will tend to be conservative at temperatures below 1900°F. Effects of internal oxidation from UO_2 and steam pressure effects are not included.	Conservative, but acceptable model is used.
Above 1900°F, Cathcart's data is acceptable.	The Cathcart correlation is used.	In conformance with Reg. Guide position.

3.2.6 Heat Transfer from Reactor Internals

Staff Position	GE Process	Evaluation
Heat transfer from piping, vessel walls and internal hardware should be calculated in a best-estimate manner.	TRACG models pipe and vessel walls as well as internal hardware as "heat slabs". Conduction through the slabs is modeled as 1-D process across the slab with radial nodalization of the walls [1, Section 4]. Geometrical complexity (at penetrations, etc.) is not simulated, but masses and surface areas of the structures are preserved. Heat transfer coefficients correspond to the fluid regimes in contact with the heat slabs. Single-phase convection to liquid or vapor, subcooled and nucleate boiling and condensation are modeled [1, Section 6.6].	Heat transfer from reactor internals is modeled in a best-estimate manner consistent with a system code representation, to assure that that heat releases to the fluid are calculated accurately. Calculations of integral experiments (TLTA, FIST, GIRAFFE/SIT) show good comparisons for pressure response and voiding in the lower plenum. The uncertainty in this parameter is largely in the value of the heat transfer coefficients. Sensitivity studies have been made on the heat transfer coefficients as part of the uncertainty study in Section 2.4.4.

3.3 Reactor Core Thermal/Physical Parameters

3.3.1 Thermal Parameters for Swelling and Rupture of the Cladding and Fuel Rods

Staff Position	GE Process	Evaluation
The model should calculate fuel cladding swelling and rupture resulting from the temperature distribution in the cladding and from the pressure difference between the inside and outside of the cladding, both as a function of time.	TRACG calculates swelling and rupture based on an empirical fit to experimental data for BWR size fuel rods. The cladding strain is a function of the cladding temperature and the hoop stress (Section 7.5.3.3 of Reference 1).	Requirements are met. TRACG model for cladding swelling is empirically based, rather than true best estimate. No fuel cladding swelling will occur in ESBWR LOCA as the core is always covered.

Staff Position	GE Process	Evaluation
The degree of swelling and rupture should be taken into account in the calculation of gap conductance, cladding oxidation and embrittlement, hydrogen generation, and heat transfer and fluid flow outside of the cladding.	The change in gap size affects the gap conductance calculation (Section 7.5.2.5 of Reference 1). Cladding oxidation and hydrogen generation are functions of the cladding surface area. Changes in cladding embrittlement are not calculated by TRACG. While the effects of the area change on the flow outside the rod can be handled by TRACG, the analysis does not account for this effect. Experimental data have shown insensitivity to this effect.	See above. Cladding embrittlement is not calculated in TRACG. Requirements for coolable geometry are met by meeting criteria on PCT and oxidation. No fuel cladding swelling will occur in ESBWR LOCA as the core is always covered.
The calculation of fuel and cladding temperatures as a function of time should use values of gap conductance and other thermal parameters as functions of temperature and time.	TRACG has a dynamic gap conductance model (Section 7.5.2 of Reference 1) which accounts for changes in gap conductance, plenum temperature, rod internal pressure and thermal properties with time.	The TRACG gap conductance model meets the requirements of the Reg. Guide.
The calculation of the swelling of cladding should take into account spatially varying cladding temperatures, heating rates, anisotropic material properties, asymmetric deformation of cladding, and fuel rod thermal and mechanical parameters.	TRACG simulates the fuel rod with axial and radial nodes. The calculation of cladding swelling accounts for spatial variations in temperatures and heating rates. Asymmetric effects are accounted for empirically through the use of data.	TRACG model for cladding swelling is empirically based and meets Reg. Guide requirements. No fuel cladding swelling will occur in ESBWR LOCA as the core is always covered.

3.3.2 Other Core Thermal Parameters

Staff Position	GE Process	Evaluation
Physical and chemical changes in in-core materials (e.g., eutectic formation, phase change, etc.) should be included as necessary.	TRACG does not model physical and chemical changes in in-core materials.	These phenomena are not significant for ESBWR LOCAs, and their treatment is not necessary.

3.4.1 Break Characteristics and Flow

Staff Position	GE Process	Evaluation
The critical flow model should consider the fluid conditions at the break location, upstream and downstream pressures, and break geometry.	The TRACG critical flow model (Section 6.3 of Reference 1) accounts for break conditions (subcooled, two-phase, steam), and upstream and downstream pressures. Break geometry can be treated with the use of discharge coefficients.	Split and double-ended breaks can be analyzed. The TRACG model is empirically based but accounts for all relevant parameters and has been shown to be accurate by extensive comparisons to data.

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The uncertainties and bias of the model should be stated, as well as the range of applicability.	The uncertainty and bias for the TRACG critical flow model have been quantified (Section 6.3.6 of Reference 1).	TRACG model meets Reg. Guide requirements.

3.4.2 ECC Bypass

Staff Position	GE Process	Evaluation
ECC bypass during the blowdown phase of a LOCA should be calculated in a best estimate manner. One-dimensional models justified through analysis and data are acceptable.	TRACG models “flooding” or CCFL type of phenomena through a Kutateladze type of correlation (Section 6.1.7.2 of Reference 1). The correlation used in TRACG is conservative for predicting ECC bypass in the downcomer (Section 6.1.7.4 of Reference 1).	The ECC bypass phenomenon is important for PWRs, but is not significant for BWRs (Section 6.1.7.4 of Reference 1). Therefore, a conservative model is acceptable for BWR analysis. Also, in the ESBWR, the GDCS flow enters the vessel at the end of blowdown.

3.5 Noding Near the Break and ECCS Injection Point

Staff Position	GE Process	Evaluation
Sufficient sensitivity studies should be performed on the noding and other important parameters to ensure calculations provide realistic results.	Sensitivity to nodalization near the break and ECC injection point has been studied. Nodalization is consistent between test facilities and ESBWR in these regions. Uncertainties in other parameters are considered as part of the PIRT parameter uncertainty study.	Process meets Reg. Guide requirements.

3.6 Frictional Pressure Drop

Staff Position	GE Process	Evaluation
The frictional pressure drop in pipes and other components should be calculated using models that include variation of friction factor with Reynolds number and effects of two-phase flow effects on friction.	Wall friction is calculated with a fit to the Moody curves as a function of Reynolds number and surface roughness (Section 6.2.1.3 of Reference 1). The two-phase multiplier is a modified Chisholm multiplier (Section 6.2.1.4 of Reference 1).	Models are in conformance with Reg. Guide requirements.
The gravitational, friction and acceleration components of pressure drop should be consistently calculated.	The terms in the phasic momentum equations are consistently formulated and calculated (Section 3 of Reference 1).	Most data comparisons are for total pressure drop. Since the void fraction is compared against other data, these comparisons are checks on the consistency of the pressure drop components.
Model should be checked against experimental data and the bias and uncertainty should be stated.	The frictional pressure drop models in TRACG have been extensively compared with experimental data for tubes and bundles (Section 6.2.1.6 of Reference 1). Estimates of the mean bias and uncertainty are also given in the same section.	Models are in conformance with Reg. Guide requirements.

3.7 Momentum Equation

Staff Position	**GE Process**	**Evaluation**
The momentum equation should include terms for: 1) temporal change in momentum, 2) momentum convection, 3) area change momentum flux, 4) momentum change due to compressibility, 5) pressure loss resulting from wall friction, 6) pressure loss resulting from area change, and 7) gravitational acceleration.	The momentum equations are formulated for each phase and contain all the relevant terms (Section 3.1.2 of Reference 1 for the differential form; Sections 3.2.1.1 and 3.2.2.1 of Reference 1 for the difference form).	Equations for separate phase flows are used with the appropriate interfacial terms.
Technical basis should be demonstrated with data and analysis.	The validity of the momentum equations is demonstrated by comparisons with pressure drop, void fraction and critical flow data (Sections 3.5, 3.1 and 3.4 of Reference 2).	The momentum equations represent best-estimate models and are adequately qualified against test data.

3.8 Critical Heat Flux

Staff Position	GE Process	Evaluation
Best-estimate models developed from appropriate steady-state or transient experimental data should be used for calculating CHF.	TRACG uses the best-estimate GEXL correlation for calculation of CHF (Section 6.6.6.1 of Reference 1). The GEXL correlation is based on an extensive database for steady-state CHF in BWR rod bundles. At low flow conditions, a modified Zuber correlation is used (Section 6.6.6.1 of Reference 1).	The correlations cover the range of LOCA conditions. The correlations have been validated for time varying conditions that exist in operational transients and LOCAs (Sections 3.2.1 and 5.1.2 of Reference 2). The boiling length correlation is known to be accurate over a large range of lengths and covers the 10 ft active core height of the ESBWR. A larger value of uncertainty (5% vs. 3.2%) is assumed for the analysis.
Return to nucleate boiling is allowed if justified by local fluid and surface conditions.	TRACG allows a return to transition boiling if the wall temperature is below T_{min} and the local quality is less than the critical quality. Nucleate boiling is restored when the wall temperature is less than T_{CHF}.	The TRACG model has been validated against test data from BWR rod bundles (Sections 5.1.2, 5.2.3 and 3.6.2 of Reference 2-11). No fuel heatup will occur in ESBWR LOCA as the core is always covered.
Technical basis should be demonstrated with data and analysis.	The TRACG CHF model has been extensively qualified for transient conditions simulating LOCAs [2].	The TRACG model is in conformance with the requirements of the Reg. Guide.

3.9 Post-CHF Blowdown Heat Transfer

Staff Position	GE Process	Evaluation
A model for post-CHF heat transfer should: a. Be checked against an acceptable set of relevant data. b. Recognize effects of liquid entrainment, thermal radiation, and thermal nonequilibrium, low and high mass flow rates, low and high power densities and saturated and subcooled inlet conditions.	TRACG calculates post-dryout heat transfer in two regimes: (1) dispersed droplet flow at high flow and qualities, and (2) inverted annular flow at low flow rates and low qualities. These heat transfer regimes are described in Sections 6.6.9 and 6.6.10 of Reference 1. Liquid entrainment is considered. The TRACG model allows for unequal temperatures for the two phases. The radiation model is described in Section 6.6.12 of Reference 1. The Bromley correlation for low quality film boiling has been compared against a range of bundle reflooding data (Section 6.6.9.3 of Reference 1).	The correlations cover the range of expected LOCA conditions. The correlations have also been validated against appropriate data. No fuel cladding heatup will occur in ESBWR LOCA as the core is always covered

Staff Position	GE Process	Evaluation
	Comparisons against data at high qualities are shown in Section 6.6.10.3 of Reference 1. Comparisons have also been made with the ORNL tests (Section 3.2.1 of Reference 2).	
Correlations for heat transfer from uncovered fuel bundles should: a. Be checked against an acceptable set of relevant data. b. Recognize the effects of radiation and of laminar, turbulent and transition flows.	The correlations used in the uncovered portion of the bundle are described in Section 6.6.10 of Reference 1. The single-phase steam correlation includes the laminar, turbulent and transition regimes. Additionally, the effects of droplets are accounted for through the Sun-Tien-Gonzalez correlation (Equation 6.6-49). The radiation heat transfer model is described in Section 6.6-12 of Reference 1.	The models are in conformance with the requirements of the Reg. Guide. Comparisons with core spray cooling data are shown in Section 6.6.10.3 of Reference 1. No fuel cladding heatup will occur in ESBWR LOCA as the core is always covered
Uncertainties and bias in the models for post-CHF heat transfer should be stated.	Applicability and uncertainty and bias in the low and high void fraction film boiling regimes are provided in Sections 6.6.9.3 and 6.6.10.3 of Reference 1.	Reg. Guide requirements have been satisfied.

3.10 Pump Modeling

Staff Position	GE Process	Evaluation
The characteristics of rotating primary system pumps should be derived from a best-estimate dynamic model that includes momentum transfer between the fluid and the rotating member, with variable speed as a function of time. The model for two-phase flow should be verified by comparison to applicable data.	The governing equations for the pump are given in Section 7.2.1 of Reference 1. The momentum equation for the pump component includes a term for the momentum transfer from the rotating member to the fluid. Homologous curves are used to characterize the pump head and torque as a function of the fluid volumetric flow and pump speed. To account for the two-phase effects on pump performance, degradation factors based on data are applied.	There are no primary system pumps in ESBWR.

3.11 Core Flow Distribution During Blowdown

Staff Position	GE Process	Evaluation
The core flow through the hottest region (no larger than one fuel bundle) should be calculated as function of time. Calculations should account for any crossflow between regions.	The high power bundle is modeled as a separate region in TRACG.	This requirement is aimed at PWR analysis. Because of the BWR configuration with zircaloy channels surrounding each bundle, there is no crossflow between bundles.

3.12 Post-Blowdown Phenomena

3.12.1 Containment Pressure

Staff Position	GE Process	Evaluation
The containment pressure used for evaluating effectiveness during the post-blowdown phase of a LOCA should be best-estimate and include the effects of containment heat sinks.	The containment is explicitly modeled for LOCA analysis and includes the effects of heat sinks in the containment. Additionally, sensitivity studies have been made with respect to containment pressure.	Reg. Guide requirements met.

3.12.2 Calculation of Post-Blowdown Thermal Hydraulics for Pressurized Water Reactors

Not Applicable

3.12.3 Steam Interaction with ECC Water in Pressurized Water Reactors

Not Applicable

3.12.4 Post-Blowdown Heat Transfer for Pressurized Water Reactors

Not Applicable

3.13 Convective Heat Transfer Coefficients for BWR Rods Under Spray Cooling

Staff Position	GE Process	Evaluation
Following the blowdown period, convective heat transfer coefficients should be determined based on the calculated fluid conditions and heat transfer modes.	TRACG applies convective heat transfer coefficients following blowdown corresponding to the calculated heat transfer regime. The heat transfer selection logic is shown in Section 6.6.2 of Reference 1.	The TRACG models have been extensively qualified [2],[24] for tests simulating jet pump BWRs and ESBWR.
During the period following lower plenum flashing, but prior to ECC initiation, heat transfer models should include steam cooling or two-phase flow convection.	TRACG applies convective heat transfer coefficients following blowdown corresponding to the calculated heat transfer regime. The heat transfer selection logic is shown in Section 6.6.2 of Reference 1. Steam cooling, nucleate boiling and film boiling are considered.	TRACG models for post lower plenum flashing heat transfer phenomena are best-estimate and meet the Reg. Guide requirements.

Staff Position	**GE Process**	**Evaluation**
Following ECC initiation, but prior to reflooding, heat transfer models should account for rod-to-rod variations in heat transfer.	Best-estimate correlations are used for steam/droplet cooling (Section 6.6.10 of Reference 1), and rod-rod, and rod-channel radiative heat transfer with an absorbing medium (Section 6.6.12 of Reference 1). TRACG spray heat transfer models have been validated against spray cooling tests (Section 6.6.10.3 of Reference 1).	TRACG has the required models. These effects are not important for ESBWR, as there is no core uncovery.
After the two-phase level reaches the level under consideration, a best-estimate heat transfer model should be used. This model should include the effects of any flow blockage.	TRACG applies convective heat transfer coefficients following blowdown corresponding to the calculated heat transfer regime. The heat transfer selection logic is shown in Section 6.6.2 of Reference 1. Typically, the modified Bromley correlation (Section 6.6.9 of Reference 1) would be used at low void fractions.	Effects of flow blockage due to swelling of cladding are not considered in TRACG other than as an increase in surface area of the fuel rod cladding. Experimental data [56]have shown minor sensitivity to even large amounts of flow blockage. No fuel cladding swelling or flow blockage will occur in ESBWR LOCA as the core is always covered and well cooled.
Thermal hydraulic models that do not consider multiple channels should be compared with experimental data or more detailed calculations to ensure that all important phenomena are adequately calculated.	Multiple channels are modeled in TRACG.	Comparison with data from the 30° Steam Sector Test Facility (Section 5.4 of Reference 2) has shown the capability of TRACG to model the multi-channel phenomena seen in the refill/reflood phase of a BWR.

3.14 BWR Channel Box Under Spray Cooling

Staff Position	GE Process	Evaluation
Following the blowdown period, heat transfer from the channel box and wetting of the channel box should be determined based on the calculated fluid conditions on both sides of the channel box and should make use of best-estimate rewetting models that have been compared with applicable experimental data.	TRACG applies convective heat transfer coefficients following blowdown corresponding to the calculated heat transfer regime on either side of the channel box. The heat transfer selection logic is shown in Section 6.6.2 of Reference 1. TRACG employs a quench front propagation correlation (Section 6.6.13 of Reference 1), which is a fit to the two-dimensional conduction solution. These models have been extensively validated against core spray cooling data[54],[2].	Reg. Guide requirements met. No fuel cladding heatup will occur in ESBWR LOCA as the core is always covered. There are no core spray systems in ESBWR.

3.15 Special Considerations for a Small-Break LOCA in Pressurized Water Reactors

Not Applicable

3.16 Other Features of Best-Estimate Codes

Staff Position	GE Process	Evaluation
Completeness: Comparisons of the overall calculations to integral experiments should be performed to ensure that important phenomena can be predicted.	Comparisons of TRACG predictions against integral experiments are shown in Section 5 of Reference 2 and in Volume 2 of Reference 24. An overall assessment of TRACG capabilities to predict this data is shown in Reference 24.	The integral test comparisons show that all major LOCA phenomena are captured by TRACG.
Data Comparisons: Individual models should be compared against data. Uncertainty and bias in models should be evaluated.	Comparisons of TRACG against separate effects data are shown in Sections 3.1 through 3.9 of Reference 2 and in Volume 1 of Reference 24.	The separate-effects test comparisons show that the individual models in TRACG predict separate-effects phenomena correctly.

Regulatory Position 4: Estimation of Overall Calculational Uncertainty

4.1 General

Staff Position	GE Process	Evaluation
The calculational uncertainty should include the uncertainty due to individual models ("code uncertainty"), experimental data, boundary and initial conditions, fuel behavior and simplifying assumptions.	Uncertainties due to individual models, boundary and initial conditions and fuel behavior are accounted for explicitly. Some boundary and initial conditions are chosen conservatively. Experimental data were selected for comparisons based on adequate accuracy in the experiments. Deviations between data and calculations implicitly include experimental uncertainties. Effects of simplifying assumptions are implicit in comparisons with integral tests.	The required uncertainty components are accounted for.
A 95% probability level is acceptable for comparing best-estimate predictions to the applicable limits of Paragraph 50.46(b) of 10CFR50.	Calculations are intended to bound the 95th percentile value of the minimum chimney static head	Meets Reg. Guide requirements.

4.2 Code Uncertainty

Staff Position	GE Process	Evaluation
[[		]]
It will be necessary to evaluate the code’s predictive ability over several time intervals.	The entire transient is considered in the evaluation rather than a single value.	Meets Reg. Guide requirements.
[[		]]

4.3 Other Sources of Uncertainty

Staff Position	GE Process	Evaluation
Uncertainties associated with boundary and initial conditions (initial power, pump performance, valve actuation times and control systems operational) should be accounted for. It is acceptable to limit the variables to conservative bounds.	Sensitivity studies have been performed to assess the effect of changes in boundary and initial conditions. Many variables have been set to conservative values (technical specification limits).	Meets Reg. Guide requirements.
Uncertainties in fuel parameters such as fuel conductivity, gap width, gap conductivity and peaking factors should be accounted for in the uncertainty analysis.	Uncertainties in the fuel conductivity, gap width and conductance are treated as individual model uncertainties contributing to the uncertainty in the fuel rod stored energy. Uncertainties in the peaking factor are included in the initial conditions.	Meets Reg. Guide requirements.

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4.4 Statistical Treatment of Overall Calculational Uncertainty

Staff Position	GE Process	Evaluation
[[		]]
Justification should be provided for the assumed parameter distributions and ranges.	Justification for the assumed parameter distributions and ranges is provided in Section 2.4.	This corresponds to the CSAU step on ranging of the parameters under Step 4.
The evaluation of PCT at the 95% level need only be performed for the limiting break. Justification must be provided that the overall calculational uncertainty at the limiting condition bounds that at the other conditions.	Calculations are performed for the limiting break. See discussion above regarding the lack of impact on PCT.	The requirements of the Reg. Guide are met.

2.1.5.2 Conformance with CSAU Methodology

The TRACG LOCA application methodology also addresses all the elements of the NRC-developed Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology. The CSAU methodology is documented in the report *Quantifying Reactor Safety Margins, Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break Loss-of-Coolant Accident* [15]. The CSAU report describes a rigorous process for evaluating the total model and plant parameter uncertainty for a nuclear power plant calculation. Further details on the CSAU methodology are contained in the NRC-issued Regulatory Guide 1.157. The CSAU methodology incorporates the elements of phenomena identification and ranking, documentation of models, assessment against Systems Effects Tests (SETs) and Integral System Tests (ISTs) for the key phenomena, and quantification of uncertainties due to the models, scaling and plant parameters. In the CSAU process, the model uncertainty is derived from the propagation of individual model uncertainties through code calculations; experimental comparisons are used as a check on the derived uncertainty. This process will be followed with TRACG, but for the ESBWR a simpler bounding approach will be used to combine uncertainties.

The CSAU methodology consists of 14 steps, as outlined in Table 2.1- 1.

Table 2.1- 1. CODE SCALING, APPLICABILITY AND UNCERTAINTY EVALUATION METHODOLOGY

CSAU Step	Description	Addressed In
1	Scenario Specification	Section 2.1.5.2
2	Nuclear Power Plant Selection	Section 2.1.5.2
3	Phenomena Identification and Ranking	Section 2.2
4	Frozen Code Version Selection	Reference [1]
5	Code Documentation	References [1,2,24,25,26]
6	Determination of Code Applicability	Section 2.3
7	Establishment of Assessment Matrix	Section 2.3.2
8	Nuclear Power Plant Nodalization Definition	Section 2.4.2
9	Definition of Code and Experimental Accuracy	Reference [1,2,24]
10	Determination of Effect of Scale	Section 2.4.3
11	Determination of the Effect of Reactor Input Parameters and State	Section 2.5
12	Performance of Nuclear Power Plant Sensitivity Calculations	Section 2.6,2.7
13	Determination of Combined Bias and Uncertainty	Section 2.6,2.7
14	Determination of Total Uncertainty	Section 2.6,2.7

The 14 CSAU steps are summarized in the following paragraphs. The objectives for each step are addressed by indicating how they will be addressed in this report.

1. **Specify scenario.**

   The LOCA scenarios include the full range of pipe breaks analyzed for the ESBWR. Further, the scenarios are differentiated for large and small breaks and by the location of the break. Typical ESBWR LOCA scenarios are described in Section 2.2.1. For LOCAs, the transient has been divided into the *Blowdown, GDCS and Long Term PCCS* phases. Of these only the first two are relevant for ECCS/LOCA considerations. The subdivision into phases allows reduction of the analysis to only those processes and components that are important during each phase.

2. Select nuclear plant.

The Nuclear Power Plant (NPP) which is the basis for this application report is the 4000 MWt ESBWR described in detail in the ESBWR Design Description [23].

3. Identify and rank phenomena.

All processes and phenomena that occur during an event do not equally influence plant behavior. The most cost efficient, yet sufficient, analysis reduces all candidate phenomena to a manageable set by identifying and ranking the phenomena with respect to their influence on the primary safety criteria. The phases of the events and the important components are investigated. The processes and phenomena associated with each component are examined. Cause and effect are differentiated. After the processes and phenomena have been identified, they are ranked with respect to their effect on the primary safety criteria for the event. A phenomena identification and ranking table (PIRT) is established to guide the subsequent uncertainty quantification. The PIRTs for ECCS/LOCA are developed in Reference 29 and reported in Section 2.2.

4. Select frozen code.

TRACG04A is the frozen code selected for the analysis. TRACG02A was the code frozen for AOO analysis. The only major model additions to create TRACG04A are the axial conduction controlled quench front model, PANAC11 physics and I/O changes. An earlier version of TRACG04 was used for the majority of the validations presented in the SBWR TRACG Qualification Report [24]. The recent additions in TRACG04A should have no impact on the earlier qualification. This has been confirmed by running spot checks on the SBWR qualification cases. All aspects of management, control, maintenance, testing and documentation of the code are governed by internal procedures (see Section 2.5.1).

5. Document code.

The details of the models are contained in the *TRACG Model Description* LTR [1]. A summary description of the TRACG assessment is provided in Section 2.3.2. Details are contained in the *TRACG Qualification* LTRs [2], [24], [25]. This report describes the application process. The User' Manual [26] provides guidance on the use of the code.

6. Determine code applicability.

To demonstrate *applicability*, one must begin with *capability*. Capability to calculate an event for a nuclear power plant rests on four elements: (1) conservation equations, which provide the code capability to address global processes; (2) constitutive correlations and models, which provide code capability to model and scale particular processes; (3) numerics, which provide code capability to perform efficient and reliable calculations; and (4) structure and nodalization, which address code capability to model plant geometry and perform efficient and accurate plant calculations. All four elements must be considered when evaluating the code capability for a specific application. Code capability is only one aspect needed to demonstrate that the code is applicable.

Applicability also implies that the capability of the code has been demonstrated by actually applying the code in the intended manner and then qualifying the results. The capability of TRACG to model phenomena that are important to ESBWR simulations has been addressed in Table 2.3-1 in Section 2.3.1. Q*ualification* aspects have also been addressed in Section 2.3.2.

7. Establish assessment matrix.

The determination of uncertainty for a computer code must be based on a sufficient data set, which necessarily will include both separate and integral effects tests and available plant data. The assessment matrix must cover all phenomena and components that were identified and ranked important in the PIRT for the selected events for the nuclear power plant. The LOCA PIRTs are documented in Section 2.2. The assessment coverage of the PIRTs is summarized in Table 2.3-2 through Table 2.3-5.

8. Define nodalization for plant calculations.

The plant model must be nodalized finely enough to represent both the important phenomena and design characteristics of the nuclear power plant but coarsely enough to remain economical. In principle, nodalization can be treated as an individual contributor to code uncertainty; however, quantification of nodalization uncertainty can be very costly. Thus, the preferred path is to establish a standard nodalization based on the assessment against separate and integral effects tests. Nodalization studies have been performed in assessing this test data in order to determine the level of detail necessary to represent the important phenomena and then consistent levels of detail have been applied to establish standard noding schemes for the ESBWR. The standard ESBWR nodalization for TRACG for ECCS/LOCA applications is defined based on the qualification and is described in *TRACG Qualification* for SBWR [24].

9. Determine code and experiment uncertainty.

Simulations against experiments are used to determine the code accuracy. Comparisons to separate effects tests are used to quantify the uncertainty in the individual models and correlations. Typically, experimental uncertainty is inherent in these comparisons and is not separated out. Quantification of the uncertainties in the model parameters is discussed in Section 2.4.1. The impact on the primary safety parameters for the nuclear power plant can be determined by varying the inputs to the individual models by a specified amount (e.g. $\pm$ 1 σ). The overall uncertainty of the code in simulating the important phenomena for ECCS/LOCA is addressed fully in Section 2.4.

10. Determine effects of scale.

The differences for similar physical processes, at scales up to and including full scale, should be evaluated to establish a statement of potential scaling effects. For TRACG, this has been done by evaluating the experimental basis for the individual models and correlations against full-scale plant conditions, by performing qualification against separate-effects tests, integral effects tests at different scales and full-scale plant data (where plant data exist), and by using a plant nodalization based on the qualification studies. Specific evaluations for ECCS/LOCA are addressed in Section 2.4.3.

11. Determine effects of plant operating conditions.

Uncertainties in the nuclear power plant simulations may result from uncertainties in plant operating state at the initiation of the LOCA or in plant process parameters. For example, the plant power distribution is a function of burnup history and control rod pattern prior to the transient. For the ESBWR, these uncertainties are accounted for by using analytical limits for parameters that influence ECCS/LOCA response (Section 2.5.3).

12. Perform plant sensitivity calculations.

Nuclear power plant calculations for a given event are used to determine the code's output sensitivity (in the primary safety criteria parameters) to various plant operating conditions that arise from uncertainties in the reactor state at the initiation of the transient event or in plant process parameters. Similarly, nuclear power plant calculations are used to address the uncertainties introduced by the code models and correlations. In this manner, the sensitivities of the safety-related quantities to these parameters are evaluated individually or collectively. The sensitivity studies for ECCS/LOCA are documented in Section 2.4.4.

13. Combine biases and uncertainties.

In this step, all the biases and uncertainties are combined into an overall bias and uncertainty. There are different techniques that can be used, as discussed in Section 2.6. Because there is no core heatup for the ESBWR for LOCAs, a bounding approach has been adopted. The results of the ECCS/LOCA analysis are shown in Section 2.7.

14. Determine total uncertainty.

The statement of total uncertainty for the code for ESBWR ECCS/LOCA analysis is given in terms of the difference between the bounding and nominal results.

2.1.6 Implementation Requirements

The implementation of TRACG into actual licensing analysis is contingent on completion of the following implementation requirements:

- Review and approval by the NRC of:

    1. The uncertainties documented in Section 2.4.

    2. The bounding process for analyzing ECCS/ LOCA described in Section 2.6.

- Analysis for the ESBWR LOCA break spectrum and the overall biases and uncertainties to be applied to the limiting LOCAs are included in this report. The acceptance criteria (PCT, local oxidation and core-wide oxidation) are automatically met as long as the core remains covered. Demonstration of core coverage is based on application of the application processes described in Section 2.6. These results demonstrate compliance with the acceptance criteria (Section 2.7).

The criteria for updating the overall bias and uncertainty in subsequent plant cycles are discussed in more detail in Section 2.7.

2.1.7 Review Requirements For Updates

In order to effectively manage the future viability of TRACG for ESBWR ECCS/LOCA licensing calculations, GE proposes the following requirements for upgrades to the code to define changes that (1) require NRC review and approval and (2) that will be on a notification basis only.

2.1.7.1 Updates to TRACG Code

Modifications to the basic models described in Reference 1 may not be made for ECCS/LOCA licensing calculations without NRC review and approval.

Changes in the numerical methods to improve code convergence may be used in ECCS/ LOCA licensing calculations without NRC review and approval, as long as the cumulative effect of these changes on the calculated PCT is less than 50^0 F. These changes will be subject to reporting under the requirements of 10CFR50.46.

Features that support effective code input/output may be added without NRC review and approval.

2.1.7.2 Updates to Fuel Rod Thermal Mechanical Model for ECCS/LOCA Application

The NRC-approved GESTR/LOCA model [27] has been used to initialize the TRACG calculations in this application report. The NRC may approve updates to the fuel rod model in the future. In this event, the updated fuel rod model may be used for the same purpose in ECCS/ LOCA licensing calculations without NRC review and approval as long as the safety parameters of PCT, local oxidation and core-wide oxidation are not impacted compared to the model used in this LTR (i.e. the core remains covered at all times). A typical ECCS/ LOCA calculation for the limiting break will be performed and the results of the comparison will be transmitted for information.

2.1.7.3 Updates to TRACG Model Uncertainties

New data may become available with which the specific model uncertainties described in Section 2.4 may be reassessed. If the reassessment results in a need to change specific model uncertainty, the specific model uncertainty may be revised for ECCS/ LOCA licensing calculations without NRC review and approval as long as the process for determining the uncertainty is unchanged. These changes will be subject to reporting under the requirements of 10CFR50.46.

2.1.7.4 Updates to TRACG Application Method

Revisions to the TRACG application method described in Section 2.6 may not be made for ECCS/LOCA licensing calculations without NRC review and approval.

2.1.7.5 Cycle Specific Uncertainties in Safety Parameters

Biases and uncertainties in the minimum two-phase level in the chimney are developed for the ESBWR plant using the process described in this report. This process will be implemented for the first operating cycle for the ESBWR. The magnitudes of these biases and uncertainties may change for future core designs and do not require NRC review and approval. The values of the uncertainties will be transmitted to the NRC for information if the margin to core uncovery is significantly impacted.

2.1.8 Range of Application

The intended application is ECCS/LOCA analysis as required by 10CFR50.46 for ESBWR. This covers the entire spectrum of break sizes and locations. The break could be initiated anywhere in the operating domain for an ESBWR operating at or below the technical specification limits. Equipment out of service or performance relaxations can also be analyzed. The application range includes, but is not restricted to:

- Initial, transition, and equilibrium cores
- ADS valve out of service
- Feedwater heater out of service
- MSIV out of service
- Feedwater temperature reduction

2.2 Phenomena Identification and Ranking

The critical safety parameters required by 10 CFR 50.46 for ECCS/ LOCA are peak cladding temperature (PCT), maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling. Additional intermediate safety parameters include the downcomer level and two-phase mixture level inside the core shroud. These safety parameters are the criteria used to judge the performance of the safety systems and the margins in the design. It is expected that only the two-phase level inside the shroud is relevant for the ESBWR because the core does not uncover for any LOCA. The values of the critical safety parameters are determined by the governing physical phenomena. To delineate the important physical phenomena, it has become customary to develop Phenomena Identification and Ranking Tables (PIRTs). PIRTs are ranked with respect to their impact on the critical safety parameters. For example, the two-phase level inside the shroud is determined by the reactor vessel inventory and inventory distribution between the various vessel regions, core power generation, core flow etc.

All processes and phenomena that occur during a LOCA do not equally influence plant behavior. The most cost efficient, yet sufficient, analysis reduces all candidate phenomena to a manageable set by identifying and ranking the phenomena with respect to their influence on the critical safety parameters. The phases of the events and the important components are investigated. The processes and phenomena associated with each component are examined. Cause and effect are differentiated. After the processes and phenomena have been identified, they are ranked with respect to their effect on the critical safety parameters for the event. The identification of important phenomena for the ESBWR was done in two ways: (1) a Top-Down process based on analyses and sensitivity studies, and (2) a Bottom-Up process based on examination of individual design features [29].

Section 2.2.1 describes representative TRACG calculations that established the scenarios of various LOCA events. The descriptions stress the phenomenological evolution of the transients. The scenarios are then reviewed by interdisciplinary teams to identify each thermal-hydraulic phenomenon that plays a role in the analysis, and to rank all of them in terms of "importance"; that is, degree of influence on the figure of merit (e.g., two-phase level inside the core shroud). Section 2.2.2 reports the results of the phenomena ranking from References 29 and 24.

2.2.1 LOCA Transient Response

Chapter 6 of the SSAR will include the entire matrix of calculations for postulated pipe rupture locations and single failures. For a complete PIRT evaluation, the entire spectrum of events must be covered, including analyses with less limiting conditions than the design-basis case with no auxiliary power. The approach followed in this study was to focus initially on the design basis cases, in terms of the equipment and systems available. This led to the most severe consequences and the greatest challenges to the analytical models in modeling the phenomena. The next step was to examine the possible interactions with other systems that might be available, even though they are not classified as engineered safeguard features for the event. To facilitate understanding, a large break in the Gravity-Driven Cooling System (GDCS) line has been chosen to illustrate the sequence of events during the LOCA. The sequence of events is similar for all the LOCA events, particularly after initiation of the GDCS flows, when the vessel and containment transients are coupled. While there are some differences in the assumptions made for analysis of the different breaks, these are not very important in determining the phenomenological progression of the LOCA or the importance of various parameters. The limiting LOCA from the perspective of margin to core uncovery is a large liquid line (GDCS line) break; from the viewpoint of containment pressure, it is likely to be the large steamline break. A schematic of the ESBWR passive safety systems is shown in Figure 2.2-1.

The overall LOCA sequence can be divided into three periods: blowdown period, GDCS period and the long-term cooling PCCS period. These periods are shown in Figure 2.2-2. The ***Blowdown period*** is characterized by a rapid depressurization of the vessel through the break, safety relief valves (SRVs) and depressurization valves (DPVs). The steam blowdown from the break and DPVs pressurizes the drywell, clearing the main containment vents and the PCCS vents. First, noncondensible gas and then steam flows through the vents and into the suppression pool. The steam is condensed in the pool and the noncondensible gas collects in the wetwell air space above the pool. At about 500 seconds, the pressure difference between the vessel and the wetwell is small enough to enable flow from the GDCS pools to enter the vessel. This marks the beginning of the ***GDCS period***, during which the GDCS pools drain their inventory. Depending on the break size and location, the pools are drained in between 2000 and 7000 seconds. The GDCS flow fills the vessel to the elevation of the break, after which the excess GDCS flow spills over into the drywell. The GDCS period is characterized by condensation of steam in the vessel and drywell, depressurization of the vessel and drywell and possible openings of the vacuum breakers, which returns noncondensible gas from the wetwell airspace to the drywell. The decay heat eventually overcomes the subcooling in the GDCS water added to the vessel and boiloff resumes. The drywell pressure rises until flow is reestablished through the PCCS. This marks the beginning of the ***Long-term PCCS cooling period***. During this period, the noncondensible gas that entered the drywell through the vacuum breakers is returned to the wetwell. Condensate from the PCCS is recycled back into the vessel through the PCCS drain tank in the drywell.

The most important part of the ECCS/LOCA transient for vessel response is the blowdown period and the early part of the GDCS period when the vessel is reflooded and inventory restored. For some breaks (e.g. bottom drain line break), the equalization line from the suppression pool to the reactor vessel may open during the long-term cooling period (after more than 24 hours) to provide the vessel an additional source of makeup water if the water level in the downcomer falls to 1m above the elevation of the top of active fuel.

2.2.1.1 GDCS Line Break

The GDCS line break scenario is a double-ended guillotine break of a GDCS drain line. There are three GDCS pools in the ESBWR containment, supplying four divisions of GDCS to the vessel. Each drain divides into two branches before entering into the pressure vessel. Each branch has a check valve followed by a squib operated injection valve and finally a nozzle in the vessel wall to control the blowdown flow in case of a break. The check valve prevents backflow from the vessel to the pool. The GDCS break is assumed to occur in one branch, between the squib-operated valve and the nozzle entering the vessel. Additional assumptions for the LOCA analysis include a simultaneous loss of auxiliary power and no credit for the on-site diesel generators. The only AC power assumed available is that from battery powered inverters.

- ***Blowdown Period*** — At break initiation, the assumed simultaneous loss of power trips the generator, causing the turbine bypass valves to open and the reactor to scram. The bypass valves close after 6 seconds. No credit is taken for this scram or the heat sink provided by the bypass. The power loss also causes a feedwater coastdown. Drywell cooling is lost and the control rod drive (CRD) pumps trip. The blowdown flow quickly increases the drywell pressure to the scram setpoint.

High drywell pressure isolates several other functions, including the Containment Atmosphere Control System (CACS) purge and vent, Fuel and Auxiliary Pool Cooling System (FAPCS), high and low conductivity sumps, fission product sampling, and reactor building Heating, Ventilating and Air Conditioning (HVAC) exhaust.

Loss of feedwater and flow out the break cause the measured water level in the downcomer to drop past the Level 3 (L3) scram setpoint. The "measured" or "sensed" downcomer level corresponds to the static head in the downcomer above the lower instrument tap used for the wide range level instrument. This setpoint will scram the reactor if it has not already scrammed on high drywell pressure. The scram will temporarily increase the rate of measured downcomer level drop and the Level 2 (L2) trip will quickly follow the L3 trip. This trip will isolate the steamlines and open the isolation condenser (IC) drain valves, but no credit is taken in the safety analysis for heat removal by the IC. After L2, the rate of decrease in the downcomer sensed level will slow and, without external makeup, the Level 1 (L1) trip will be reached, but not for several minutes. During this delay, the IC, if available, would be removing energy and reducing pressure and break flow. After a 10-second delay to confirm the L1 condition, the Automatic Depressurization System (ADS) logic will start a timed sequential opening of depressurization and injection valves. Four SRVs (one on each steamline) open first. The remaining eight SRVs open in two stages to stagger SRV line clearing loads in the suppression pool and minimize downcomer level swell. Similarly, opening of the depressurization valves (DPVs) is delayed 45 seconds. Two DPVs on the main steamlines open first, followed by two stages of two additional DPVs. The remaining two DPVs open after an additional delay. Blowdown through the break and the SRVs and DPVs causes a level swell in the downcomer and chimney, which collapses at the end of the blowdown period, with the GDCS injection. Ten seconds after the last DPV opens, the GDCS injection valves are opened. When the GDCS injection valves first open, the hydrostatic head from the pool plus the wetwell pressure (GDCS pools are located in the wetwell) is not sufficient to open the check valves and GDCS flow does not begin immediately. When the GDCS check valves do open, the cold GDCS water further depressurizes the vessel.

GDCS Period — The GDCS flow begins refilling the vessel and the downcomer two-phase level rises. When the two-phase level reaches the break, the GDCS flow spills back into the drywell. For the GDCS break, the flow of GDCS water is sufficient to raise the downcomer two-phase level above the break, until the pools empty, then the level drains back to the break elevation. Inside the core shroud, the two-phase level in the chimney also decreases after depressurization, but is restored after the GDCS refills the vessel. Figure 2.2-3 shows the expected chimney and downcomer two-phase levels during the first 2000 s of the transient. The two-phase level swell during the initial blowdown and opening of the SRVs and DPVs is not visible in the figure (note the level drop and then rise during the GDCS period as the vessel is refilled).

For the GDCS break, the reactor core does not uncover, so there is no cladding heatup above the initial operating temperature. In evaluating the "importance" of various phenomena in the PIRT process, the phenomena associated with cladding heatup (e.g., radiation heat transfer, metal-water reaction) are unimportant, while phenomena associated with the two-phase level inside the core shroud (e.g., decay heat, energy release from heat slabs) are comparatively important.

The LOCA scenario develops slowly for the ESBWR. The accident detection system logic functions almost instantaneously, but thereafter, the time scales are measured in hours rather than seconds. The chimney two-phase level (Figure 2.2-3) dips briefly about 10 minutes into the LOCA due to void collapse following GDCS injection. For the GDCS line break, the minimum chimney level (> 1 m above the top of the core) occurs at about 10 to 12 hours after the break. At this point in time, the core void fraction is very small, and the chimney and downcomer levels are almost the same. This slow response, which is due to the large volume of water in the reactor vessel and GDCS pools, makes the LOCA a very slow moving event from the reactor systems and operator response standpoint.

For the ECCS/LOCA transient response, the primary interaction with the containment is in the determination of the GDCS initiation time. The wetwell pressure will also decrease as the GDCS pools drain, thus slowing down the rate of injection slightly. The minimum two-phase level in the chimney occurs shortly after the GDCS starts to inject. Subsequently, there is no effect of the containment boundary conditions on the ECCS/LOCA transient.

2.2.1.2 Main Steamline Break

In this subsection, the important features of the transient resulting from a large break in the main steamline are described. The emphasis is on those features that are different from the GDCS line break scenario.

- ***Blowdown Period*** — At break initiation, the blowdown flow quickly increases the drywell pressure to the scram setpoint, and a control rod scram occurs. The high velocities in the steamline initiate closure of the Main Steamline Isolation Valves (MSIVs) and the reactor isolates in 3 - 5 seconds. This trip also opens the Isolation Condenser (IC) drain valves, but no credit is taken in the safety analysis for heat removal by the IC. High drywell pressure isolates several other systems, including the Containment Atmosphere Control System (CACS) purge and vent, Fuel and Auxiliary Pool Cooling System (FAPCS), high and low conductivity sumps, fission product sampling, and reactor building Heating, Ventilating and Air Conditioning (HVAC) exhaust.

  Loss of feedwater and flow from the break cause the vessel water level to drop. Without external makeup, the Level 1 (L1) trip will be reached in about 6 minutes. During this period, the IC, if available, would be removing energy and reducing pressure and break flow. After a 10-second delay to confirm the L1 condition, the Automatic Depressurization System (ADS) logic starts a timed sequential opening of depressurization and injection valves. The SRVs open in several stages to stagger SRV line clearing loads in the suppression pool and to minimize vessel level swell. The sequence of opening of the DPVs and the GDCS injection valves is similar to that for the GDCS line break described earlier. However, because of the large steam break, the vessel depressurizes faster and GDCS injection begins earlier than for the GDCS line break. Blowdown through the break, the SRVs, and the DPVs causes a level swell in the vessel. The two-phase level in the downcomer decreases at the end of the blowdown period, when GDCS injection begins.

- ***GDCS Period*** **—** The GDCS flow begins refilling the vessel and the downcomer two-phase level rises. When the two-phase level reaches the elevation of the open DPVs, the GDCS flow spills back into the drywell. Inside the core shroud, the two-phase level in the chimney also decreases after depressurization, but is restored after the GDCS refills the vessel. The minimum two-phase level in the chimney is of the order of 3 m above the top of the core; there is substantial margin to core heatup.

2.2.1.3 Small Breaks

The thermal hydraulic phenomena that characterize the small breaks in the ESBWR are very similar to those for the large steamline break. This is because once the downcomer level drops below the Level 1 set point, the reactor is automatically depressurized through the SRVs and DPVs. For small breaks (depending on the size and location), it may take several minutes before the reactor is scrammed on low water level (Level 3), and still longer before the ADS is actuated. For a steamline break having an area equivalent to 2% of the main steamline cross-sectional area, the measured downcomer water level will boil off to reach Level 1 in about one hour. During this period, the break flow exceeds the condensing capacity of the PCCS and results in clearing the top row of horizontal vents. This results in energy addition to the portion of the suppression pool above the top vents, and increases the pool surface temperatures. The ESBWR incorporates an ADS trip on high suppression pool surface temperature in conjunction with a high drywell pressure to mitigate this effect.

2.2.1.4 Non-Design Basis LOCAs

The discussion to this point has focused on LOCA scenarios with design basis assumptions. The consequences of relaxing these assumptions towards a “best estimate scenario” and considering the availability of non-safety systems are examined in this subsection.

Single Failures:

In the ESBWR, the active component failures considered are the failure of a valve in the GDCS line to open and the failure of a DPV to open. Scenarios without failures have been analyzed. With no failures, design margins are increased. No new thermal-hydraulic phenomena or interactions are introduced because the differences relate simply to the number of GDCS lines available (quantity of GDCS flow) or the number of DPVs available for depressurization (amount of steam blowdown flow and rate of depressurization). Tests with both types of single failure and ones without any failure were included in the LOCA simulations performed in the GIST facility.

Isolation Condenser Operation:

For LOCA analysis, the IC is not treated as an engineered safety feature and no credit is taken in the safety analysis for its operation. The valve in the condensate return line will open in a realistic scenario. This increases the vessel liquid inventory before ADS and reduces the steam load on the containment. LOCA scenarios with the IC operational have been included in the consideration of important phenomena in Sections 2.1.1 through 2.1.3. These phenomena include the IC condensation efficiency, steam quenching in the reactor vessel downcomer, and interactions between the IC steam flow and the steam flow through the DPVs on the same nozzle.

Diesel Generators Available:

Additional non-safety systems become available when the diesel generators start up (Table 2.2-1). Only the Control Rod Drive System in its high-pressure injection mode is initiated automatically. This system injects water through the feedwater line into the downcomer. The Fuel and Auxiliary Pool Cooling System (FAPCS) will also be available to the operator with the diesels operational. FAPCS isolates automatically on high drywell pressure. The operator can override the isolation manually. The FAPCS has several modes of operation. It can be aligned to function initially in the Low Pressure Coolant Injection (LPCI) mode. When core cooling is established, the FAPCS can serve as a Suppression Pool cooling system. Interactions between the FAPCS and the passive safety systems (GDCS/PCCS) are uniformly beneficial and increase LOCA margins [29].

Offsite Power Available:

Table 2.2-2 shows that the primary additional water makeup systems available with offsite power are the condensate and feedwater systems. Numerous auxiliary systems such as fuel pool cooling, drywell coolers, and drywell sump drain pumps would also be available. With feedwater and offsite power available, the accident becomes a relatively mild event. After scram on high drywell pressure, the feedwater maintains normal downcomer water level for an extended period of time even for large breaks. This allows the operator to initiate a controlled depressurization of the reactor. The water spilling out of the reactor collects in the lower drywell. For large breaks, the sump drain pumps will not be able to keep up with the break discharge. Eventually, water spills into the wetwell through the spillover holes in the pipes connected to the horizontal vents. The feedwater will be throttled back or turned off as the water level rises in the wetwell.

2.2.2 Composite List of Identified Phenomena and Interactions

The composite list of highly ranked phenomena and interactions for ECCS/LOCA primarily considers single failure scenarios and those with the Isolation Condenser available. Multiple failures have been excluded. A more detailed explanation of what the phenomena are and the basis for the judgment on their relative importance is provided in the ESBWR TAPD and Supplement 1 of the TAPD report [29].

Table 2.2-3 is a list of highly ranked phenomena for ECCS/ LOCA. A relatively large number of phenomena in this table are "generic"; that is, common for all BWRs.

While the base LOCA scenario does not claim credit for the Isolation Condenser, the Isolation Condenser can be expected to operate and have a beneficial effect on the transient by retaining vessel inventory during the blowdown phase. Because each Isolation Condenser unit consists of two modules coming off a single riser, and as many as four units could be in operation, interactions between modules and units are possible (XL1). The interaction between the system depressurization rate and GDCS affects GDCS timing and the minimum liquid inventory during the transient. This interaction has been designated XL3. It is a subset of Interaction XL4, which is the integral system response of the reactor vessel and containment during the late blowdown period, assuming the Isolation Condensers are available.

Table 2.2-1. LOCA Scenario with Diesel Generators Available - Additional Systems Functional

Symptom	Action(s)
Loss of normal AC	Diesel Generator starts
	FMCRD run-in backs up hydraulic scram
Low water level L2	CRD initiates in high pressure injection mode
Above actions are automatic, no operator action necessary. Actions below require operator intervention.	
Low water level L3	FAPCS LPCI mode, injection through FW system
High pool temperature	FAPCS Pool cooling mode, if adequate core cooling. Operator action required to over-ride system isolation.
Low water level < L1 per EPG	External water source
Containment pressure high or T dw > Technical Specifications LCO	DW Cooler
GDCS Pool level < NWL - 0.5m (2 of 3 pools)	Trip CRD pumps

Table 2.2-2. LOCA Scenario with Offsite Power & Diesel Generators Available

Symptom	Action(s)
Low water level L3	FW and condensate injection
Pressure > normal setpoint	Turbine bypass valves

Table 2.2-3. Composite List of Highly Ranked Phenomena for ECCS/LOCA

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Table 2.2-3 Composite List of Highly Ranked Phenomena for ECCS/LOCA (Continued)

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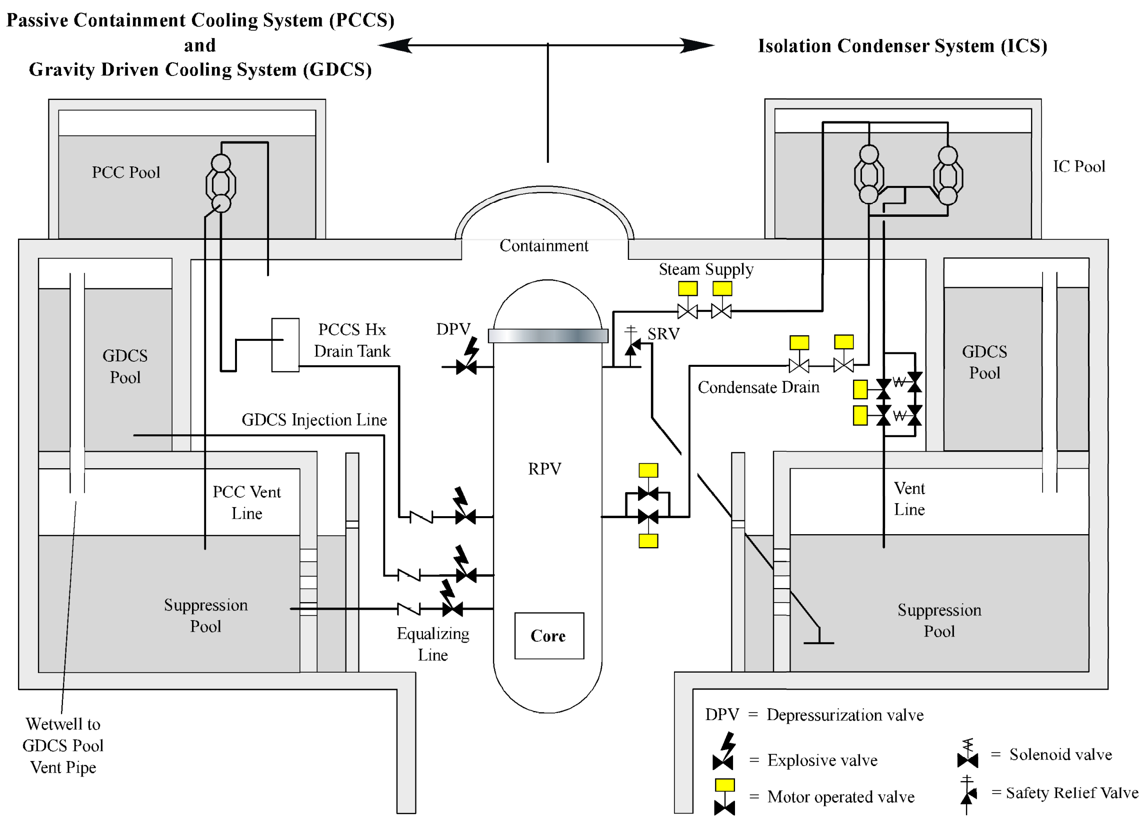

Figure 2.2-1 ESBWR Passive Safety Features

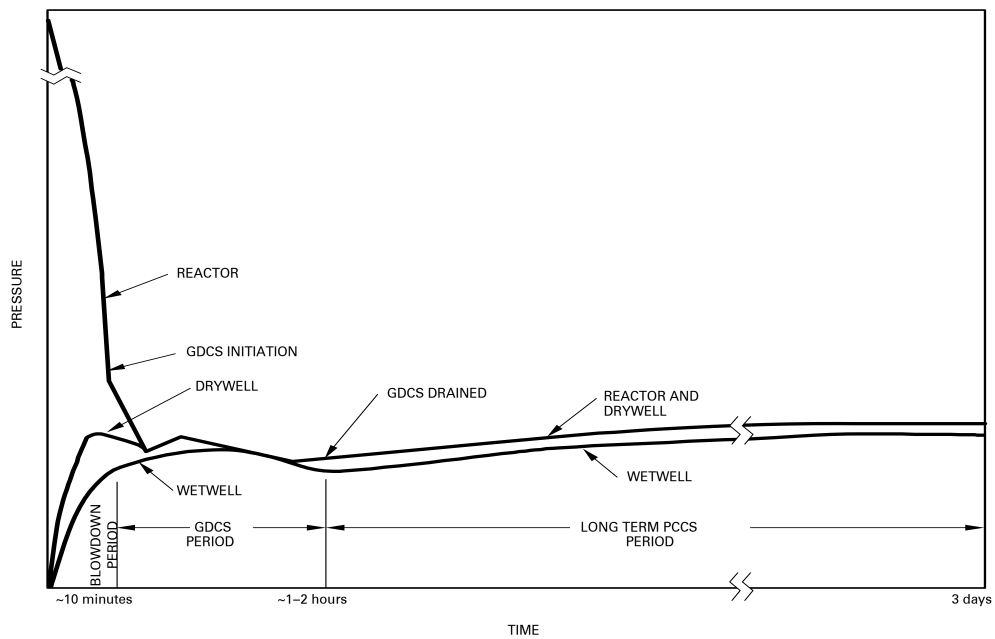

Figure 2.2-2 Phases of the LOCA Transient

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Figure 2.2-3. GDCS Line Break - Chimney and Downcomer Two-Phase Levels vs. Time

2.3 Applicability of TRACG to ESBWR ECCS/ LOCA

The objective of this section is to demonstrate the applicability of TRACG for the analysis of LOCAs in ESBWR. To accomplish this purpose, the capability of the TRACG models to treat the highly ranked phenomena and the qualification assessment of the TRACG code for ECCS/ LOCA applications is examined in the next two subsections.

2.3.1 Model Capability

The capability to calculate an event for a nuclear power plant depends on four elements:

- Conservation equations, which provide the code capability to address global processes.
- Correlations and models, which provide code capability to model and scale particular processes.
- Numerics, which provide code capability to perform efficient and reliable calculations.
- Structure and nodalization, which address code capability to model plant geometry and perform efficient and accurate plant calculations.

Consequently, these four elements must be considered when evaluating the applicability of the code to the event of interest for the nuclear power plant calculation. The key phenomena for each event are identified in generating the PIRTs for ECCS/LOCA application, as indicated in Section 2.2. The capability of the code to simulate these key phenomena is specifically addressed, documented, and supported by qualification in References 2 and 24.

Important BWR phenomena have been identified and TRACG models have been developed to address these phenomena as indicated in Table 2.3-1. For each model, the relevant elements from the Model Description LTR [1] are identified. The Interactions listed in Table 2.2-3 have not been included in Table 2.3-1 because the calculation of system interactions does not involve any new models beyond those needed for the individual phenomena. Table 2.3-1 shows that TRACG has models for all the highly ranked phenomena for ECCS/LOCA.

2.3.2 Model Assessment Matrix

For each of the governing BWR phenomena, TRACG qualification has been performed against a wide range of data. In this section, the qualification basis is related to the phenomena that are important for ECCS/LOCA. This is a necessary step to confirm that the code has been adequately qualified for the intended application.

The list of highly ranked phenomena for ECCS/LOCA is cross-referenced to the qualification basis. Data from separate effects tests (Table 2.3-2), component tests (Table 2.3-3), integral system tests (Table 2.3-4) and plant data (Table 2.3-5) have been used to qualify the capability of TRACG to model the phenomena. The tables show that highly ranked phenomena for ESBWR ECCS/LOCA are well covered by TRACG assessment.

Table 2.3-1
HIGH RANKED ESBWR ECCS/LOCA PHENOMENA AND TRACG MODEL CAPABILITY MATRIX

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Table 2.3-1 High Ranked ESBWR ECCS/LOCA Phenomena and TRACG Model Capability Matrix (Continued)

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Table 2.3-2

Separate Effects Tests for Highly Ranked Phenomena for TRACG Qualification for ESBWR – ECCS/LOCA

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Table 2. 3-2 (continued)
Separate Effects Tests for Highly Ranked Phenomena for TRACG Qualification for ESBWR – ECCS/LOCA

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Table 2.3-3

Component Tests for Highly Ranked Phenomena for

TRACG Qualification for ESBWR – ECCS/LOCA

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Table 2.3-3 (cont'd)
Component Tests for Highly Ranked Phenomena for
TRACG Qualification for ESBWR – ECCS/LOCA

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Table 2.3-4

Integral System Tests for Highly Ranked Phenomena for TRACG Qualification for ESBWR – ECCS/LOCA

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						]]

Table 2.3-4 (cont'd)
Integral System Tests for Highly Ranked Phenomena for
TRACG Qualification for ESBWR – ECCS/LOCA

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Table 2.3-5

BWR Plant Data for Highly Ranked Phenomena for TRACG Qualification for ESBWR – ECCS/LOCA

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Table 2.3-5 (cont'd)
BWR Plant Data for Highly Ranked Phenomena for
TRACG Qualification for ESBWR – ECCS/LOCA

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2.4 Model Uncertainties and Biases

Model biases and uncertainties for LOCA application of TRACG are assessed as described below for each of the high ranked phenomena identified in Section 2.2. The assessments are typically performed on the basis of comparisons between separate effects test data and TRACG calculations performed with the best-estimate version of the code. The biases and uncertainties indicated by the data comparisons are used to establish probability density functions (PDFs) for TRACG parameters and correlations. These are implemented into TRACG through special input parameters designated as "PIRT multipliers". The correspondence between the PIRT multiplier inputs and the models they modify is shown in Table 2.4-3. Biases are compensated by appropriate choice of the mean value of the PIRT multiplier and uncertainties are accommodated by choosing PDFs to represent the standard deviation of the data comparisons. In general, no attempt is made to separate out the uncertainty in the data comparisons for the possible effect of measurement errors; i.e. measurement uncertainties are implicitly included in the standard deviation of the data comparisons. There are some parameters affecting the high ranked phenomena for which no applicable test data are available. For these cases, the PIRT model uncertainty is chosen on the basis of engineering judgment and comparisons with similar parameters for which data are available. In some instances, the parameter was found to have little impact on the figure of merit for the LOCA calculation (e.g., two-phase level inside the core shroud) and it was possible to use a conservative estimate of the uncertainty. The results of this evaluation are summarized in Table 2.4-1.

2.4.1 Model Parameters and Uncertainties

This section discusses the biases and uncertainties in the TRACG parameters and correlations that have a potential effect on each of the high-ranked phenomena listed in Table 2.2-3. As in Table 2.2-3, the presentation is organized by plant region, starting with the lower plenum and ending with the steamline. Under the heading of each phenomenon, the applicable TRACG parameters and correlations are identified, the sources of the test data and the statistical characteristics of the deviations between TRACG calculations and the test data are described and the choice of the PDF is explained. The results of the evaluation are summarized in Table 2.4-1. In addition, the sensitivity of the calculated mixture level inside the shroud is discussed in Section 2.4.4.2.

Lower Plenum

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Table 2.4-1. Parameters Governing High Ranked PIRT Phenomena

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Table 2.4-2. Summary of Scoping Break Spectrum Analysis – Minimum Static Head inside Chimney

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Table 2.4-3. TRACG PIRT Parameters ranged for ECCS/LOCA Sensitivity Study

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Figure 2.4-1 Void Fraction Deviations for Tests Applicable to Regions with Large Hydraulic Diameter

Figure 2.4-2. Sensitivity of TRACG Prediction of Average Void Fraction in EBWR Test Facility to PIRT Multiplier on Interfacial Drag Coefficient

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Figure 2.4-3. Probability Distribution for Multiplier on Interfacial Drag Coefficient

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Figure 2.4-4. Void Fraction Deviations for Toshiba Void Fraction Tests

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Figure 2.4-5. Sensitivity of TRACG Prediction of Toshiba Void Fraction to PIRT Multiplier on (C_o-1)

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Figure 2.4-6. Sensitivity of TRACG Prediction of Toshiba Void Fraction PIRT Multiplier on Entrainment Coefficient, η

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Figure 2.4-7. Fractional Error in Modified Zuber Critical Heat Flux Correlation

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Figure 2.4-8. Comparison of the Predicted and Measured Two-Phase Level Histories for Marviken Test 24

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Figure 2.4-9. Comparison of the Predicted and Measured Two-Phase Level Histories for Marviken Test 15

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Figure 2.4-10. Deviation in Level Change Versus the Hydraulic Diameter for Separate Effects and Integral Facilities

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Figure 2.4-11. GDCS Line Break with GDCS Injection Valve Failure – Two-phase Level inside Chimney

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Figure 2.4-12. GDCS Line Break with GDCS Injection Valve Failure – Collapsed Level (Static Head) inside Chimney

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Figure 2.4-13. Chimney Static Head Sensitivity to Uncertainties in TRACG PIRT Parameters (See Table 2.4-3)

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Figure 2.4-14. PCT Sensitivity to Uncertainties in TRACG PIRT Parameters (See Table 2.4-3)

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Figure 2.4-15. Chimney Static Head Sensitivity to TRACG Simulation of DW Noncondensible Holdup

2.5 Application Uncertainties and Biases

2.5.1 Input

Specific inputs for ECCS/LOCA calculations are specified via internal procedures, which are the primary means used by GE to control application of engineering computer programs. The specific code input will be developed in connection with the application LTR and the development of the application specific procedure. This section will be limited to a more general discussion of how input is treated with respect to quantifying the impact on the calculated results. As such, it serves as a basis for the development of the application specific procedures.

Code inputs can be divided into four broad categories: (1) geometry inputs; (2) model selection inputs; (3) initial condition inputs; and (4) plant parameters. For each type of input, it is necessary to specify the value for the input. If the calculated result is sensitive to the input value, then it is also necessary to quantify the uncertainty in the input.

The geometry inputs are used to specify lengths, areas and volumes. Uncertainties in these quantities are due to measurement uncertainties and manufacturing tolerances. These uncertainties usually have a much smaller impact on the results than do other uncertainties associated with the modeling simplifications. When this is not the case, the specific uncertainties can usually be quantified in a straightforward manner.

Individual geometric inputs are the building blocks from which the spatial nodalization is built. Another aspect of the spatial nodalization includes modeling simplifications such as the lumping together of individual elements into a single model component. For example, several similar fuel channels may be lumped together and simulated as one fuel channel group. An assessment of these kinds of simplifications, along with the sensitivities to spatial nodalization, is included in the *qualification reports* [2], [24].

Model selection inputs are used to select the features of the model that apply for the intended application. Once established, these inputs are fully specified in the procedure for the application and will not be changed.

A distinction has been made in this document between *initial conditions* and *plant parameters*. Obviously, when specified in absolute units, the initial rated conditions for a nuclear power plant are specific to the plant and thus have been considered as plant parameters in some documents. In this document, *initial conditions* are considered to be those key plant inputs that determine the overall steady-state nuclear and hydraulic conditions prior to the transient. These are inputs that are essential to determining that the steady-state condition of the plant has been established.

The name *plant parameter*, on the other hand, is reserved for such things as protection system setpoints, valve capacities that influence the characteristics of the transient response but which do not (when properly prescribed) have an impact on steady-state operation.

2.5.2 Plant Conditions Used for Base Line Calculations

Based on prior experience, it is assumed for design basis ECCS/LOCA analyses that the preferred electric power is lost simultaneously with the initiation of LOCA. As a further conservatism, the ESBWR design analyses do not take credit for non-ECCS vessel inventory control systems including, specifically, the Feedwater System, the Isolation Condenser System and the Control Rod Drive system. The significant plant input variables used for the base line ECCS analyses are given in Table 2.5-1.

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Table 2.5-1. Significant Input Variables to the Loss-of-Coolant Accident Analysis

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Table 2.5- 1. Significant Input Variables to the Loss-of-Coolant Accident Analysis (Continued)

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Table 2.5-2. Plant Variables with Nominal And Sensitivity Study Values

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Figure 2.5-1. Chimney Static Head Sensitivity to Plant Parameter Uncertainties

2.6 Combination of Uncertainties

In order to determine the total uncertainty in predictions with a computer code, it is necessary to combine the uncertainties due to model uncertainties (CSAU Step 9), scaling uncertainties (CSAU step 10), and plant condition or state uncertainties (CSAU Step 11). Various methods have been used to combine the effects of uncertainties in safety analysis. All these approaches are within the framework of the CSAU methodology, since the CSAU methodology does not prescribe the approach to be used.

NRC Regulatory Guide 1.157 for use of best-estimate models for LOCA analysis defines acceptable model features and application procedures. The guide states that a one-sided upper statistical limit (OSUSL) can be calculated at the 95% probability level for the primary safety parameters. In addition, the statistical methodology should be provided and justified.

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2.7 Results for ECCS/LOCA Analysis

In this Section, TRACG results are presented for ECCS/LOCA analysis for the ESBWR. The results include:

1. Nominal TRACG analyses for the limiting break,

2. Bounding analysis in accordance with the process defined in Section 2.6.1.

2.7.1 Nominal ECCS/LOCA Analysis

A baseline analysis was performed for the GDCS line break with a failure of one GDCS injection valve to open. This was determined to be the limiting LOCA in Section 2.4.4. The plant initial conditions are specified in Table 2.5-1.

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Figure 2.7-1. TRACG Nodalization of ESBWR RPV and containment for ECCS/LOCA Analysis

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Figure 2.7-2. TRACG Nodalization of ESBWR Steam Line System

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Figure 2.7-3. TRACG Nodalization of ESBWR IC, DPV and Feedwater Systems

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Figure 2.7-4. RPV Pressure Response (Base Case)

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Figure 2.7-5. RPV, Drywell and Wetwell Pressure Response (Base Case)

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Figure 2.7-6. Two-Phase Levels in Downcomer and Inside Core Shroud (Base Case)

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Figure 2.7-7. Two-Phase Level and Static Head In Chimney (Base Case)

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Figure 2.7-8. RPV Pressure Response (Bounding Case)

[[ ]]

Figure 2.7-9. RPV, Drywell and Wetwell Response (Bounding Case)

[[ ]]

Figure 2.7-10. Two-Phase Levels in Downcomer and Inside Core Shroud (Bounding Case)

[[ ]]

Figure 2.7-11. Two-Phase Level and Static Head In Chimney (Bounding Case)

2.8 Summary of ECCS/LOCA Application Methodology

This report has defined an application methodology that meets the licensing requirements for ECCS /LOCA analysis for the ESBWR.

The requirements to be met and the scope of application were identified in Sections 2.1. Phenomena important for ECCS/LOCA analysis for ESBWR were identified in Section 2.2. Section 2.3 justified the applicability of TRACG for ECCS/LOCA analysis. Model and plant parameters and their ranges were established in Sections 2.4 and 2.5. A bounding application approach was proposed in Section 2.6. Results with this bounding approach were presented for the limiting GDCS line break in Section 2.7 and shown to have large margin to core uncovery.

Hence, conformance to design limits such as PCT and oxidation is assured.

3. CONTAINMENT/LOCA ANALYSIS

3.1 Licensing Requirements and Scope of Application

The NRC Standard Review Plan, NUREG-0800 [19], presents the responsibilities and guidelines for the NRC's reviews of nuclear power plants. The sections of the Standard Review Plan (SRP) that are relevant to the TRACG analysis for the ESBWR are Section 6.2.1 covering the containment functional design. The specific elements of these sections of the SRP that are relevant to ESBWR applications of TRACG are presented in this section of this report. These guidelines, in general, require the use of methods that have been reviewed and approved by the NRC. The TRACG Model Description NEDE-32176P, TRACG Qualification NEDE-32177, TRACG Qualification for SBWR Volumes 1 and 2, NEDC-32725, and TRACG Qualification for ESBWR NEDC-33080 are incorporated by reference as part of the review scope.

3.1.1 Licensing Acceptance Criteria for Containment/LOCA Performance

The NRC guidelines for review of Containment/LOCA safety analysis are identified in Section 6.2.1, *Containment Functional Design*, of the SRP [19]. Two statements from the introduction of this section relate directly to the TRACG analyses of the ESBWR Containment/LOCA response:

- "The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line or feedwater line break accidents."

- "GDC (General Design Criteria) 50, among other things, requires that consideration be given to the limitations in defining accident phenomena, and the conservatism of calculational models and input parameters, in assessing containment design margins."

Guidelines which are more specific to BWR pressure suppression containments are identified in SRP Section 6.2.1.1.C, *Pressure-Suppression Type BWR Containments.* Although this section of the SRP covers Mark I, II, and III pressure-suppression containments, it has been used as the basis for the review of the ABWR containment safety analysis by the NRC. The ESBWR containment design has evolved from the Mark III and ABWR containments. Therefore, these guidelines can be considered as the applicable basis for the review of the ESBWR containment analysis. The following statements from SRP Section 6.2.1.1.C are quoted directly in an attempt to summarize the NRC's review approach and requirements as they relate to ESBWR Containment/LOCA pressure and temperature response analysis using TRACG.

I. AREAS OF REVIEW

"1. The temperature and pressure conditions in the drywell and wetwell due to a spectrum (including break size and location) of loss-of-coolant accidents."

"5. The capability of the containment to withstand the effects of steam bypassing the suppression pool."

"7. The effectiveness of static {ESBWR Passive Containment Cooling System} and active {not relevant to ESBWR} heat removal systems."

"12. The evaluation of analytical models used for containment analysis."

II. ACCEPTANCE CRITERIA

CSB {Containment Systems Branch of NRC} accepts the containment design if the relevant requirements of General Design Criteria 4, 16, 38, 50, and 53 are complied with. The relevant requirements are as follows:

GDC 16 and 50, as they relate to the containment being designed with sufficient margin, require that the containment and its associated systems can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

Specific criterion or criteria that pertain to design and functional capability of BWR pressure-suppression type containments are indicated below:

If an analytical model other than the General Electric Mark III analytical model is used, the model should be demonstrated to be physically appropriate and conservative to the extent that the General Electric model has been found acceptable. In addition, it will be necessary to demonstrate its performance with suitable test data in a manner similar to that described above.

For Mark III plants at the construction permit stage, containment design pressure should provide at least a 15% margin above the peak calculated containment pressure, and the design differential pressure between drywell and containment should provide at least a 30% margin above the peak calculated differential pressure.

GDC 38 requires that a Containment Heat Removal system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

The other containment acceptance criteria are related to missile and pipe whip protection (GDC 4), periodic inspections (GDC 53), containment dynamic loads, allowable bypass leakage rates, design leakage rate, containment negative pressures, external pressures, SRV in-plant tests, local suppression pool temperature limits during SRV discharges, and instrumentation for post-accident monitoring. These criteria are not relevant to this TRACG application method since they are addressed by other analytical methods and/or procedures.

3.1.2 Analysis Requirements

The calculational framework used for evaluating the containment systems in terms of pressure and temperature behavior is called an evaluation model. It includes one or more computer programs, the mathematical models used, the assumptions and correlations included in the program, the procedure for selecting and treating the program input and output information, the specification of those portions of the analysis not included in computer programs, the values of parameters, and all other information necessary to specify the calculation procedure. The evaluation model must comply with the acceptance criteria for Containment/LOCA described in Section 3.1.1. The evaluation model must have been previously documented and reviewed and approved by the NRC staff.

3.1.3 Standard Review Plan (SRP) Guidelines (NUREG 800)

The NRC guidelines for review of LOCA Containment safety analysis are identified in Section 6.2.1 of the SRP [19], covering the containment functional design.

3.1.4 Proposed Application Methodology

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3.1.5 Implementation Requirements

The implementation of TRACG into actual licensing analysis is contingent on completion of the following implementation requirements:

- Review and approval by the NRC of:
  - The TRACG models used for containment analysis
  - The bounding process for analyzing containment/LOCA described in Section 3.6.
- Analysis for the ESBWR LOCA break spectrum that demonstrates compliance with the acceptance criteria (Section 3.7)

3.1.6 Review Requirements For Updates

In order to effectively manage the future viability of TRACG for ESBWR Containment/LOCA licensing calculations, GE proposes the following requirements for upgrades to the code to define changes that (1) require NRC review and approval and (2) that will be on a notification basis only.

3.1.6.1 Updates to TRACG Code

Modifications to the basic models described in Reference 1 may <u>not</u> be made for containment/LOCA licensing calculations without NRC review and approval.

Changes in the numerical methods to improve code convergence may be used in containment/LOCA licensing calculations without NRC review and approval, as long as differences in the results are less than 5% in design margin.

Features that support effective code input/output may be added without NRC review and approval.

3.1.6.2 Updates to TRACG Application Method

Revisions to the TRACG application method described in Section 3.6 may not be made for containment/LOCA licensing calculations without NRC review and approval.

3.1.7 Range of Application

The intended application is containment/LOCA analysis as required by Chapter 6 of the SAR for ESBWR. This covers the entire spectrum of break sizes and locations. The break could be initiated anywhere in the operating domain for an ESBWR operating at or below the technical specification limits.

3.2 Phenomena Identification and Ranking

The critical safety parameters for containment/LOCA are the peak pressures and temperatures in the drywell and wetwell of the containment. These safety parameters are the criteria used to judge the performance of the safety systems and the margins in the design. The values of the critical safety parameters are determined by the governing physical phenomena. To delineate the important physical phenomena, it has become customary to develop Phenomena Identification and Ranking Tables (PIRTs). PIRTs are ranked with respect to their impact on the critical safety parameters. For example, the pressure inside the wetwell is determined by the blowdown flow, noncondensible transport from the drywell, suppression pool stratification, and PCCS heat removal.

All processes and phenomena that occur during a LOCA do not equally influence containment behavior. The most cost efficient, yet sufficient, analysis reduces all candidate phenomena to a manageable set by identifying and ranking the phenomena with respect to their influence on the critical safety parameters. The phases of the events and the important components are investigated. The processes and phenomena associated with each component are examined. Cause and effect are differentiated. After the processes and phenomena have been identified, they are ranked with respect to their effect on the critical safety parameters for the event. The identification of important phenomena for the ESBWR was done in two ways: (1) a Top-Down process based on analyses and sensitivity studies, and (2) a Bottom-Up process based on examination of individual design features [29].

Section 3.2.1 describes representative TRACG calculations that established the scenarios of various LOCA events. The descriptions stress the phenomenological evolution of the transients. The scenarios are then reviewed by interdisciplinary teams to identify each thermal-hydraulic phenomenon that plays a role in the analysis, and to rank all of them in terms of "importance"; that is, degree of influence on the figure of merit (e.g., wetwell pressure). Section 3.2.2 reports the results of the phenomena ranking from References 29 and 24.

3.2.1 LOCA Transient Response

Chapter 6 of the SSAR will include the entire matrix of calculations for postulated pipe rupture locations and single failures. For a complete PIRT evaluation, the entire spectrum of events must be covered, including analyses with less limiting conditions than the design-basis case with no auxiliary power. The approach followed in this study was to focus on the design basis cases, in terms of the equipment and systems available. This led to the most severe consequences and the greatest challenges to the analytical models in modeling the phenomena. To facilitate understanding, a large break in the Gravity-Driven Cooling System (GDCS) line and a large break in a main steamline have been chosen to illustrate the sequence of events during the LOCA. The sequence of events is similar for all the LOCA events, particularly after initiation of the GDCS flows, when the vessel and containment transients are coupled. While there are some differences in the assumptions made for analysis of the different breaks, these are not very important in determining the phenomenological progression of the LOCA or the importance of various parameters. The limiting LOCA from the perspective of margin to core uncovery is a large liquid line (GDCS line) break; from the viewpoint of containment pressure, it is likely to be the large steamline break.

The overall LOCA sequence can be divided into three periods: blowdown period, GDCS period and the long-term cooling PCCS period. These periods are shown in Figure 2.2-2. The ***Blowdown period*** is characterized by a rapid depressurization of the vessel through the break, safety relief valves (SRVs) and depressurization valves (DPVs). The steam blowdown from the break and DPVs pressurizes the drywell, clearing the main containment vents and the PCCS vents. First, noncondensible gas and then steam flows through the vents and into the suppression pool. The steam is condensed in the pool and the noncondensible gas collects in the wetwell air space above the pool. At about 500 s, the pressure difference between the vessel and the wetwell is small enough to enable flow from the GDCS pools to enter the vessel. This marks the beginning of the ***GDCS period***, during which the GDCS pools drain their inventory. Depending on the break, the pools are drained in between 1 and 6 hours. The GDCS flow fills the vessel to the elevation of the break, after which the excess GDCS flow spills over into the drywell. The GDCS period is characterized by condensation of steam in the vessel and drywell, depressurization of the vessel and drywell and possible openings of the vacuum breakers, which returns noncondensible gas from the wetwell airspace to the drywell. The decay heat eventually overcomes the subcooling in the GDCS water added to the vessel and boiloff resumes. The drywell pressure rises until flow is reestablished through the PCCS. This marks the beginning of the ***Long-term PCCS cooling period***. During this period, the noncondensible gas that entered the drywell through the vacuum breakers is returned to the wetwell. Condensate from the PCCS is recycled back into the vessel through the PCCS drain tank in the drywell.

The most important part of the LOCA transient for the vessel response is the blowdown period and the early part of the GDCS period when the vessel is reflooded and inventory restored. For some breaks (e.g. bottom drain line break), the equalization line from the suppression pool to the reactor vessel may open during the long-term cooling period to provide the vessel an additional source of makeup water if the water level in the downcomer falls to 1m above the elevation of the top of active fuel. For the containment, the blowdown phase determines the initial pressurization. During the GDCS phase the pressure levels off and decreases as the GDCS first shuts off steaming from the vessel and later spills over into the drywell, condensing steam in the drywell. At the end of the GDCS phase, noncondensibles that returned to the drywell because of vacuum breaker openings are returned to the wetwell gas space, and the PCCS assumes the decay heat load.

3.2.1.1 Containment Response for the GDCS Line Break

Containment response calculations assume loss of all AC power except that available from battery powered inverters, reactor power at 102% of rated power and no credit for IC operation. The single failure used is the failure to open a squib valve in one of the GDCS pool drain lines. Initial conditions are containment normal operating pressure and temperature, with the suppression pool at its maximum allowable operating temperature.

- ***Blowdown Period*** — The blowdown for the GDCS line break occurs from the vessel side of the broken line. The break flow is initially a liquid blowdown, and after the downcomer two-phase level falls below the GDCS line elevation, the break becomes a vapor blowdown. The

ADS, activated by the measured downcomer level, opens the SRVs and the DPVs. The flashing liquid (and later, steam) entering the drywell increases its pressure, opening the main containment vents and the PCCS vents. Most of the drywell noncondensible gas is swept through the main vents, the suppression pool and into the wetwell airspace. The steam flow through the vents is condensed in the suppression pool. During the blowdown phase of the transient, the majority of the blowdown energy is transferred into the suppression pool through the main vents. Within the pool, temperature stratification occurs, with the blowdown energy being absorbed primarily in the region above the open vents. The increase in drywell pressure also establishes flow through the PCCS, which absorbs part of the blowdown energy. After the DPVs have opened, the GDCS squib valves open about 150 s following the L1 signal. This causes the pool side of the broken line to drain the inventory of the one affected GDCS pool into the containment. The check valve keeps the vessel from blowing down through the unbroken branch of the GDCS line. For the GDCS break, this period of the accident lasts less than 10 minutes. The peak containment pressure in the short term is primarily set by the compression of the noncondensibles initially in the drywell into the wetwell vapor space. The controlling parameters are the ratio of the drywell to wetwell vapor volumes, and the temperature at the top of the suppression pool, which sets the steam partial pressure.

- ***GDCS Period* —** Once the vessel pressure drops below the pressure on the GDCS pool side of the check valves in the unbroken GDCS lines, the GDCS pools begin to empty their inventory into the vessel. The subcooled GDCS water quenches the core voids, stopping the steam flow from the vessel. The GDCS flow refills the vessel to the elevation of the break and then spills over into the drywell. Spillover from the break into the drywell begins at about 20 minutes into the accident and continues throughout the GDCS period of the accident. Once the GDCS flow begins, the drywell pressure peaks and begins to decrease. The decrease in drywell pressure stops the steam flow through the PCCS and main vents. The drop in drywell pressure is sufficient to open the vacuum breakers between the drywell and the wetwell airspace several times. As the GDCS pools empty, the effective wetwell gas space volume increases because the GDCS pools are connected to the wetwell gas space. The containment pressure is thereby reduced. Once the GDCS flow begins to spill from the vessel into the drywell, the drywell pressure drops further and additional vacuum breaker openings occur. Some of the noncondensible gas in the wetwell airspace is returned to the drywell through the vacuum breakers. The GDCS period of the transient continues until the GDCS pools empty and the decay heat is able to overcome the subcooling of the GDCS inventory in the vessel. Then, the drywell pressure rises and flow is re-established through the PCCS. The PCCS heat removal capacity, even while recycling noncondensible gas back to the wetwell, is sufficient to handle the steam generated by decay heat and the main vents are not reopened. Any uncondensed steam condenses and deposits its latent heat in the portion of the suppression pool above the outlet of the PCCS vent. This period of the accident is expected to last approximately 1 to 2 hours for the GDCS line break.

- ***Long-Term PCCS Period*** — After the drywell pressure transient initiated by the GDCS flow is over, the drywell pressure settles out, slightly above the wetwell airspace pressure. A drywell-to-wetwell pressure difference is established which is sufficient to open the PCCS vent and drive the steam generated by decay heat through the PCCS. By between 6 to 8 hours, the PCCS heat removal increases to nearly equal the decay heat power. During this final period of the transient, drywell pressure may rise slowly. This results from a slow increase in the wetwell airspace pressure, due to the assumed leakage flow between the drywell and wetwell airspace. Without the leakage, the containment pressure remains nearly constant or decreases slightly during the long-term period of the transient.

The LOCA scenario develops slowly for the ESBWR. The accident detection system logic functions almost instantaneously, but thereafter, the time scales are measured in hours rather than seconds. Containment response is gradual, with substantial margin to the design pressure even 72 hours after the break. This slow response permits well-considered, deliberate operator actions.

3.2.1.2 Main Steamline Break

In this subsection, the important features of the transient resulting from a large break in the main steamline are described. The emphasis is on those features that are different from the GDCS line break scenario.

- ***Blowdown Period*** — At break initiation, the blowdown flow quickly increases the drywell pressure to the scram setpoint, and a control rod scram occurs. The high velocities in the steamline initiate closure of the Main Steamline Isolation Valves (MSIVs) and the reactor isolates in 3 - 5 s. This trip also opens the Isolation Condenser (IC) drain valves, but no credit is taken in the safety analysis for heat removal by the IC. High drywell pressure isolates several other systems, including the Containment Atmosphere Control System (CACS) purge and vent, Fuel and Auxiliary Pool Cooling System (FAPCS), high and low conductivity sumps, fission product sampling, and reactor building Heating, Ventilating and Air Conditioning (HVAC) exhaust.

  Loss of feedwater and flow from the break cause the vessel water level to drop. Without external makeup, the Level 1 (L1) trip will be reached in about 6 minutes. During this period, the IC, if available, would be removing energy and reducing pressure and break flow. After a 10-second delay to confirm the L1 condition, the Automatic Depressurization System (ADS) logic starts a timed sequential opening of depressurization and injection valves. The SRVs open in several stages to stagger SRV line clearing loads in the suppression pool and to minimize vessel level swell. The sequence of opening of the DPVs and the GDCS injection valves is similar to that for the GDCS line break described earlier. However, because of the large steam break, the vessel depressurizes faster and GDCS injection begins earlier than for the GDCS line break. Blowdown through the break, the SRVs, and the DPVs causes a level swell in the vessel. The two-phase level in the downcomer decreases at the end of the blowdown period, when GDCS injection begins.

In the containment, the steam entering the drywell increases its pressure, opening the main containment vents and sweeping most of the drywell noncondensible gas through the main vents, through the suppression pool, and into the wetwell airspace. (Depending on the location of the break, a substantial portion of the noncondensibles in the lower drywell region may remain in that region and bleed out slowly later in the transient). During the blowdown phase of the transient, the majority of the blowdown energy is transferred into the suppression pool by condensation of the steam flowing through the main vents. The increase in drywell pressure causes flow through the PCCS, which also absorbs part of the blowdown energy. The ADS, activated by the measured downcomer level, opens the SRVs and the DPVs and augments the steam flow to the suppression pool and drywell, respectively. This period of the accident lasts less than 10 minutes.

- ***GDCS Period*** — The GDCS flow begins refilling the vessel and the downcomer two-phase level rises. When the two-phase level reaches the elevation of the open DPVs, the GDCS flow spills back into the drywell. Inside the core shroud, the two-phase level in the chimney also decreases after depressurization, but is restored after the GDCS refills the vessel. The minimum two-phase level in the chimney is of the order of 3 m above the top of the core; there is substantial margin to core heatup.

  Quenching of voids in the core by the GDCS flow reduces the steam outflow from the vessel to the drywell. Once the GDCS flow begins, the drywell pressure peaks and begins to decrease. Figure 3.2-1 shows the RPV, drywell and wetwell pressure response for the first 12 hours of the accident. The decrease in drywell pressure stops the steam flow through the PCCS and main vents. This pressure decrease may be sufficient to open the vacuum breakers between the drywell and the wetwell airspace. Draining of the GDCS pools helps to reduce the containment pressure as more wetwell volume becomes available for the noncondensibles in the wetwell gas space. Once GDCS flow begins to spill from the vessel into the drywell, the drywell pressure drops further and additional vacuum breakers may open. If the vacuum breakers open, some of the noncondensible gas in the wetwell airspace will return to the drywell through the vacuum breakers. The GDCS period of the transient continues until the water level in the GDCS pools equalizes with the collapsed level in the downcomer of the reactor pressure vessel and the decay heat is able to overcome the subcooling of the GDCS inventory in the vessel. Then, the drywell pressure rises and flow is re-established through the PCCS. The PCCS heat removal capacity, even while recycling noncondensible gas back to the wetwell, is sufficient to transfer the steam generated by decay heat without reopening the main vents. This period of the accident is expected to last for less than one hour. Figure 3.2-2 shows the PCCS heat removal during the first 12 hours of the transient. Also shown is the decay heat.

- ***Long-Term PCCS Period*** — After the drywell pressure transient initiated by the GDCS flow is over, the drywell pressure settles out, slightly above the wetwell airspace pressure. The Main Steamline break is the limiting break in terms of containment pressure and temperature, as most of the noncondensibles are swept out from the drywell into the wetwell in the initial blowdown phase. This part of the containment transient is similar to that for the GDCS line break. However, unlike the GDCS line break, the steam generated by the decay heat is condensed and all of it is returned to the vessel via the PCCS Drainage Tank. Thus, there is no long-term drop in the downcomer and chimney water level due to boiloff. A larger amount of water inventory is retained inside the vessel and a smaller amount in the lower drywell.

3.2.1.3 Small Breaks

The thermal hydraulic phenomena that characterize the small breaks in the ESBWR are very similar to those for the large steamline break. This is because once the downcomer level drops below the Level 1 set point, the reactor is automatically depressurized through the SRVs and DPVs. For small breaks (depending on the size and location), it may take several minutes before the reactor is scrammed on low water level (Level 3), and still longer before the ADS is actuated. For a steamline break having an area equivalent to 2% of the main steamline cross-sectional area, the measured downcomer water level will boil off to reach Level 1 in about one hour. During this period, the break flow exceeds the condensing capacity of the PCCS and results in clearing the top row of horizontal vents. This results in energy addition to the portion of the suppression pool above the top vents, and increases the pool surface temperatures. The ESBWR incorporates an ADS trip on high pool surface temperature in conjunction with high drywell pressure to mitigate this effect.

3.2.2 Composite List of Highly Ranked Phenomena and Interactions

Table 3.2-1 shows the Phenomena Identification and Ranking Table (PIRT) that was developed for ESBWR Containment/LOCA analysis.

The short-term drywell pressure response is governed by energy deposition by break flow and DPV discharge flow (DPV1 in Table 3.2-1). Energy removal from the drywell is through main vent (MV1) and PCCS flow (PC1), and condensation on walls and internal structures. The pressure difference required for clearing of the main vents controls the initial pressure increase in the drywell. Energy deposition in the wetwell is through the main vent flow (WW1), and flow through the SRV quenchers (WW2) and PCC vent lines (WW3). Thermal stratification of the suppression pool (WW6) is a key factor in determining how this energy is distributed within the pool; it sets the pool surface temperature and, therefore, the temperature and steam partial pressure in the wetwell gas space.

Another key parameter controlling the short-term wetwell pressure is the extent to which the noncondensibles (nitrogen) initially in the drywell are purged to the wetwell in the initial blowdown (DW3). The design of the containment must also account for the hydrodynamic loads due to pool swell, SRV line air clearing, condensation oscillations and chugging (TRACG is not used in the design process for this purpose. Empirical models are employed, which are based on extensive test data).

The long-term containment response is controlled primarily by the heat removal by the PCCS (PC2 and PC3). The ability of the PCCS to purge noncondensibles and its performance in the presence of noncondensibles are key issues (PC2 and PC5). The rates of drywell and wetwell energy addition and removal become progressively smaller in the long-term transient. The energy deposition in the wetwell is due to the PCC vent flow and any steam leakage from the drywell that bypasses the PCCS (DWB1).

Energy removal from the wetwell is through heat transfer in the gas space (at the pool interface and walls) and condensation on the wetwell walls (WW4 and WW5). The PCCS performance may be affected by the noncondensible distributions in the drywell (DW3). Overcooling of the drywell by the PCCS or by cold water spillover from the RPV can result in the drywell pressure falling below the wetwell pressure. Cold water could be added by flow from a broken GDCS line or spillover from the break after the GDCS fills the RPV to the break elevation (DW4). This will cause the vacuum breakers to open, bringing noncondensibles back to the drywell (VB1). The interaction between the RPV and the containment (RPV2) has been included in Table 3.2-1.

Table 3.2-1. Highly Ranked PIRT Phenomena for ESBWR Containment/ LOCA

Governing Phenomena for Containment/LOCA					
			Containment / LOCA (Focus: Containment Pressure)		
		Phase	Blowdown	GDCS	Long Term
	Phenomena	Basic Phenomena	Rank (H=High, M=Medium)		
BR	**Region: break (focus: energy addition to drywell)**				
BR1	Mass Flow	Critical Flow	H		
		Friction	H		
		Entrainment	H		
MV	**Region: main vent (focus: energy addition to suppression pool)**				
MV1	Mass flow	Void fraction/entrainment	H		
		Friction	H		
MV3	Vent clearing time		H		
SQ	**Region: SRV quenchers (focus: energy addition to suppression pool)**				
SQ1	Mass Flow	Void fraction entrainment	H		
		Critical flow	H		
		Friction	H		
DW	**Region: drywell (focus: pressure, temperature, noncondensible distribution)**				
DW1	Flashing/evaporation	Interfacial heat transfer	H		
DW2	Heat sources/sinks	Condensation	H		
		Degradation of condensation	H		
DW3	3-D effects	Phase separation	H		
		Noncondensible stratification	H	H	H
		Buoyancy/natural circulation			H
DW4	Condensation on reactor outflows	Condensation		H	

Table 3.2-1 (Contd.)

Highly Ranked PIRT Phenomena for ESBWR Containment/ LOCA

Governing Phenomena for Containment/LOCA					
			Containment/ LOCA (Focus: Containment Pressure)		
		Phase	**Blowdown**	**GDCS**	**Long Term**
	Phenomena	**Basic Phenomena**	**Rank (H=High, M=Medium)**		
WW	**Region: wetwell (focus: pressure, pool and gas temperature**				
WW1	Condensation/evaporation of main vent discharge	Interfacial heat transfer	H		
		Degradation by non-condensibles	H		
WW2	Condensation/evaporation of SRV discharge	Interfacial heat transfer	H		
WW3	Condensation/evaporation of PCC vent discharge	Interfacial heat transfer	H		H
		Degradation by non-condensibles			H
WW4	Free surface condensation/evaporation	Interfacial heat transfer			H
		Degradation by non-condensibles			H
WW5	Heat sources/sinks	Condensation	H		H
WW6	Pool mixing and stratification	Stratification/thermal plumes	H		H
WW7	3-D effects in gas space	Mixing, entrainment into jets	H		H
		Buoyancy/natural circulation			H
		Stratification of noncondensibles			H
PC	**Region: PCCS (focus: energy removal)**				
PC1	Mass flow into PCC	Friction			H
PC2	Condensation on primary side	Interfacial heat transfer		H	H
		Degradation by N/C		H	H
		Shear enhancement		H	H
PC3	Secondary side heat transfer	Natural circulation			H
PC5	Parallel PCC unit effects	Friction			H
		Void fraction			H

Table 3.2-1 (Contd.)

Highly Ranked PIRT Phenomena for ESBWR Containment/ LOCA

Governing Phenomena for Containment/LOCA					
			Containment/ LOCA (Focus: Containment Pressure)		
		Phase	Blowdown	GDCS	Long Term
	Phenomena	Basic Phenomena	Rank (H=High, M=Medium)		
PC8	Purging of noncondensibles			H	H
DWB	**Region: DW/WW Boundary**				
DWB1	Leakage	Friction			H
VB	**Region: vacuum breakers (focus: noncondensible distribution)**				
VB1	Mass flow	Friction			H
EQ	**Region: equalizing line**				
EQ1	Equalizing line mass flow	Friction			H
RPV	**Region: reactor pressure vessel (focus: steam flow/energy addition to drywell)**				
RPV2	RPV steam generation		H		
DPV	**Region: depressurization valves (focus: energy addition to drywell)**				
DPV1	Mass flow	Critical flow	H		
XC	**Interaction**				
XC2	Potential system interaction: IC/PCC units, GDCS etc.			H	H
XC6	Light noncondensibles DW/PCCS/WW				H
XC7	Early containment response (DW, WW, MV)		H		
XC8	Interaction PCC/MV		H	H	H

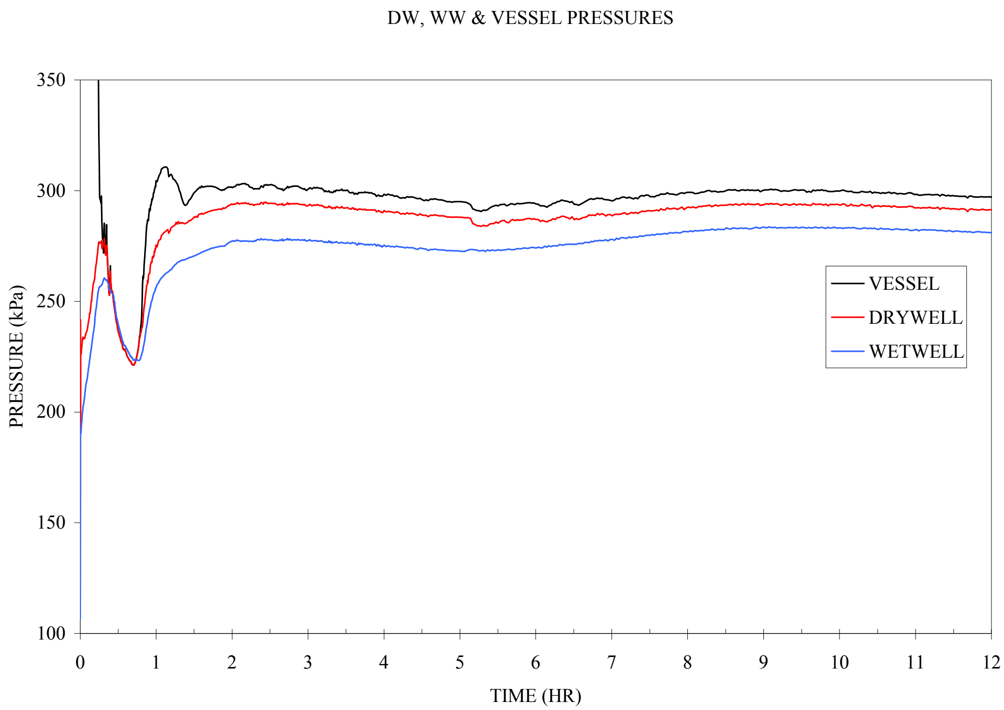

Figure 3.2-1 : Main Steam Line Break Vessel and Containment Pressures (Typical)

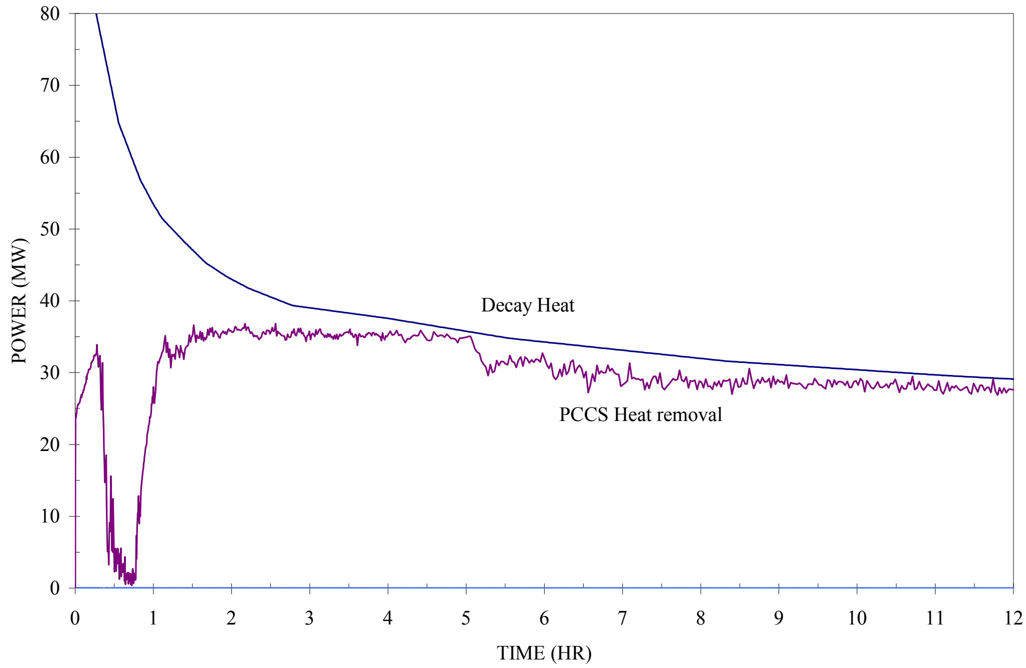

Figure 3.2-2 Main Steam Line Break Decay Heat and PCCS Heat Removal (Typical)

3.3 Applicability of TRACG to Containment/LOCA

The objective of this section is to demonstrate the applicability of TRACG for the analysis of LOCAs in ESBWR. To accomplish this purpose, the capability of the TRACG models to treat the highly ranked phenomena and the qualification assessment of the TRACG code for containment/ LOCA applications is examined in the next two subsections.

3.3.1 Model Capability

The capability to calculate an event for a nuclear power plant depends on four elements:

- Conservation equations, which provide the code capability to address global processes.
- Correlations and models, which provide code capability to model and scale particular processes.
- Numerics, which provide code capability to perform efficient and reliable calculations.
- Structure and nodalization, which address code capability to model plant geometry and perform efficient and accurate plant calculations.

Consequently, these four elements must be considered when evaluating the applicability of the code to the event of interest for the nuclear power plant calculation. The key phenomena for each event are identified in generating the PIRTs for containment/LOCA application, as indicated in Section 3.2.2. The capability of the code to simulate these key phenomena is specifically addressed, documented, and supported by qualification in References 2 and 24.

Important BWR containment phenomena have been identified and TRACG models have been developed to address these phenomena as indicated in Table 3.3-1. For each model, the relevant elements from the Model Description LTR [1] are identified. The Interactions listed in Table 3.2-1 have not been included in Table 3.3-1 because the calculation of system interactions does not involve any new models beyond those needed for the individual phenomena. Table 3.3-1 shows that TRACG has models for most highly ranked phenomena for containment/LOCA. The remaining phenomena are treated in a bounding way in the TRACG models as detailed below.

3.3.1.1 Phenomena Treated with a Bounding Approach

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3.3.2 Model Assessment Matrix

For each of the governing BWR phenomena, TRACG qualification has been performed against a wide range of data. In this section, the qualification basis is correlated to the phenomena that are important for containment/LOCA. This is a necessary step to confirm that the code has been adequately qualified for the intended application.

The list of highly ranked phenomena for containment/LOCA is cross-referenced to the qualification basis. Data from separate effects tests (Table 3.3-2), component tests (Table 3.3-3), and integral system tests (Table 3.3-4) have been used to qualify the capability of TRACG to model the phenomena. The tables show that TRACG has been adequately qualified for the calculation of ESBWR containment phenomena.

Table 3.3-1
HIGH RANKED ESBWR CONTAINMENT/LOCA PHENOMENA AND TRACG MODEL CAPABILITY MATRIX

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Table 3.3-2

Separate Effects Tests for Highly Ranked Phenomena

for TRACG Qualification for ESBWR - Containment

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Table 3.3-2 Separate Effects Tests for Highly Ranked Phenomena for TRACG Qualification for ESBWR – Containment (continued)

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Table 3.3-3
Component Tests of Highly Ranked Phenomena for
TRACG Qualification for ESBWR - Containment

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Table 3.3-3 Component Tests of Highly Ranked Phenomena for TRACG Qualification for ESBWR - Containment (Continued)

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Table 3.3-4

Integral System Tests for Highly Ranked Phenomena for

TRACG Qualification for ESBWR - Containment

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Table 3.3-4 Integral System Tests for Highly Ranked Phenomena for TRACG Qualifications for ESBWR - Containment (Continued)

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Table 3.3-5. Effect of Break Discharge Location on the Containment Pressure

	Location of the Break Discharge into the DW	Peak Drywell Pressure (psia)
Baseline Case *	Middle of Level 10 **	47.7
Sensitivity Study Case 1	Top of Level 8	44.6
Sensitivity Study Case 2	Top of Level 6	44.6
Sensitivity Study Case 3	Top of Level 1	44.0

* Baseline case described in Section 3.7.2

** See Figure 3.7-1, TRACG nodalization for ESBWR containment analysis.

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Figure 3.3-1. Wetwell Gas Space and Pool Showing TRACG Nodalization

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Figure 3.3-2. Suppression Pool Temperatures With and Without Forced Stratification (SBWR)

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Figure 3.3-3. Wetwell Gas Space Temperatures Without Forced Stratification (SBWR)

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Figure 3.3-4 -Wetwell Gas Space Temps – Restricted Mixing between Top Layer and Lower Layer (SBWR)

3.4 Model Uncertainties and Biases

Model biases and uncertainties for containment application of TRACG are assessed as described below for the key high ranked phenomena identified in Section 3.2. The assessments are typically performed on the basis of comparisons between separate effects test data and TRACG calculations performed with the best-estimate version of the code. The biases and uncertainties indicated by the data comparisons are used to establish ranges for TRACG parameters and correlations. These ranges are implemented through special inputs designated as "PIRT multipliers". Correspondence between these input parameters and the phenomena that they affect is shown in Table 3.4-1. Biases are compensated by appropriate choice of the mean value of the PIRT multiplier and uncertainties are accommodated by choosing probability density functions (PDFs) to represent the standard deviation of the data comparisons. In general, no attempt is made to separate out the uncertainty in the data comparisons for the possible effect of measurement errors; i.e. measurement uncertainties are implicitly included in the standard deviation of the data comparisons. There are some parameters affecting the high ranked phenomena for which no applicable test data are available. For these cases, the PIRT uncertainty is chosen on the basis of engineering judgment and comparisons with similar parameters for which data are available. In some instances, the parameter was found to have little impact on the figure of merit for the containment calculation (e.g., containment pressure) and it was possible to use a conservative estimate of the uncertainty. For several key parameters bounding models are used as described in Section 3.3.1.1. The results of this evaluation are summarized in Table 3.4-1.

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Table 3.4-1. Uncertainties in Highly Ranked PIRT Parameters for Containment/LOCA

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Figure 3.4-1 Comparison of PANDA Test M3 Wetwell Airspace Temperature with TRACG Predictions for WW1 Pressure.

3.5 Plant Parameters and Ranges for Application

3.5.1 Input

Specific inputs for containment/LOCA calculations are specified via internal procedures, which are the primary means used by GE to control application of engineering computer programs. The specific code input will be developed in connection with the application LTR and the development of the application specific procedure. This section will be limited to a more general discussion of how input is treated with respect to quantifying the impact on the calculated results. As such, it serves as a basis for the development of the application specific procedures.

Code inputs can be divided into four broad categories: (1) geometry inputs; (2) model selection inputs; (3) initial condition inputs; and (4) plant parameters. For each type of input, it is necessary to specify the value for the input. If the calculated result is sensitive to the input value, then it is also necessary to quantify the uncertainty in the input.

The geometry inputs are used to specify lengths, areas and volumes. Uncertainties in these quantities are due to measurement uncertainties and manufacturing tolerances. These uncertainties usually have a much smaller impact on the results than do other uncertainties associated with the modeling simplifications. When this is not the case, the specific uncertainties can usually be quantified in a straightforward manner.

Individual geometric inputs are the building blocks from which the spatial nodalization is built. Another aspect of the spatial nodalization includes modeling simplifications such as the lumping together of individual elements into a single model component. For example, several similar main vent pipes may be lumped together and simulated as one pipe. An assessment of these kinds of simplifications, along with the sensitivities to spatial nodalization, is included in the *qualification reports* [2], [24].

Model selection inputs are used to select the features of the model that apply for the intended application. Once established, these inputs are fully specified in the procedure for the application and will not be changed.

A distinction has been made in this document between *initial conditions* and *plant parameters*. Obviously, when specified in absolute units, the initial rated conditions for a nuclear power plant are specific to the plant and thus have been considered as plant parameters in some documents. In this document, *initial conditions* are considered to be those key plant inputs that determine the overall steady-state nuclear and hydraulic conditions prior to the transient. These are inputs that are essential to determining that the steady-state condition of the plant has been established.

The name *plant parameter*, on the other hand, is reserved for such things as protection system setpoints and valve capacities that influence the characteristics of the transient response but which do not (when properly prescribed) have an impact on steady-state operation. No plant parameters are important for this study.

3.5.2 Plant Initial Conditions Used for Base Line Calculations

The plant operating conditions represent initial conditions for the TRACG calculations and affect the long-term containment response. Initial conditions have an important effect on the calculated response of the containment. The range of allowable initial conditions is governed by plant operating guidelines and, for containment response calculations, it is assumed that the plant will be operated within these guidelines. In a typical calculation, initial conditions in the containment are assumed to be at steady-state, and at limiting pressures and temperatures. The RPV is assumed to be operating at maximum power and, for a given feedwater flow and temperature, the RPV steam flow, the initial temperatures and pressures and vessel internal flows are selected to obtain steady state conditions. Initial RPV power is set at 100% of rated power for the baseline calculation. Experience with similar BWR containment systems have shown that rated power produces the most limiting containment response. The only exception is a break from hot standby, which is typically included in a containment response evaluation. For this accident, it is assumed that the plant was at full power operation, is scrammed and isolated and the suppression pool is heated by SRV operation to the maximum pool temperature limit before the break occurs. This break can, for some plants, be limiting because of the high initial pool temperature. Because of the availability of the IC system following reactor isolation for the ESBWR, this break is not a concern. This is because the RPV can be depressurized without added heat load to the suppression pool.

The initial plant conditions that affect the containment response are summarized in Table 3.5-1. Some plant conditions were varied for the bounding calculation while others were maintained at nominal conditions. The basis for selection of the plant conditions to vary is discussed below.

3.5.2.1 Plant Initial Conditions Not Varied

Plant conditions that provided bounding initial conditions for the containment/LOCA analysis or conditions that would not be expected to change with normal plant operation were not varied. They included:

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3.5.2.2 Plant Conditions ranged to a Bounding Value for Sensitivity Studies

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Table 3.5-1

Plant Initial Conditions Considered in the Containment Sensitivity Study

No.	Plant Parameter	Nominal Value	Bounding Value
1	RPV Power	100%	102%
2	WW relative humidity	100%	100%
3	PCC pool level	4.8m	4.8m
4	PCC pool temperature	110F (316.5K)	110F (316.5K)
5	DW Pressure	14.7 psia (101.3kPa)	16.0 psia (110.3kPa)
6	DW Temperature	115F (319.3K)	115F (319.3K)
7	WW Pressure	14.7 psia (101.3kPa)	16.0 psia (110.3kPa)
8	WW Temperature	110F (316.5K)	110F (316.5K)
9	Suppression pool Temp.	110F (316.5K)	110F (316.5K)
10	GDCS pool temperature	110F (316.5K)	110F (316.5K)
11	Suppression pool level	5.45m	5.50m
12	GDCS pool level	6.70m	6.75m
13	DW relative humidity	20%	20%
14	RPV pressure	1040 psia (7.17 MPa)	1055 psia (7.274 MPa)
15	RPV Water Level	NWL	NWL+0.3m

3.6 Application Procedure for Containment Analysis

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3.7 Results for ESBWR Main Steamline Break LOCA

The main steamline break causes the fastest pressurization of the ESBWR drywell in the short term. It results in minimum drain-down of the GDCS pools because of the elevation of the break, and hence a smaller wetwell gas space volume in the long term. The steamline break discharging at the top of the drywell also results in a slower clearing out of the noncondensibles in the lower drywell, resulting a degraded PCCS for a longer time. All these factors lead to the highest containment pressure for the main steamline break.

3.7.1 TRACG Nodalization for Containment Analysis

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Figure 3.7-1. TRACG Nodalization for ESBWR Containment Analysis

3.7.2 Baseline Results for Containment Analysis

The RPV and containment were initialized at the base conditions shown in the Nominal Value column of Table 3.5-1. Four PCCs are available with a total rated capacity of 54 MW. A crud thickness is assumed on the tube walls corresponding to a design basis fouling factor of 0.000045 m^2-K/W or an equivalent additional inconel wall thickness of 0.65 mm (Section 3.4.1). No credit is assumed for the ICs. A leakage path was assumed between the drywell and wetwell with an equivalent area of 1 cm^2.

Apart from the conservative modeling assumptions common to all TRACG containment analysis (suppression pool stratification, wetwell gas space stratification and a break location at the top of the drywell), the other models were set at the mean values of the ranges shown in Table 3.4-1.

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Figure 3.7-2. Containment Pressure Response (Base Case)

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Figure 3.7-3. Drywell Noncondensible Partial Pressures (Base Case)

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Figure 3.7-4. 3 PCC Pool Level (Base Case)

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Figure 3.7-5. GDCS Pool Level (Base Case)

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Figure 3.7-6. PCCS Heat Removal vs. Decay Heat (Base Case)

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Figure 3.7-7. Suppression Pool Temperatures (Base Case)

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Figure 3.7-8. Wetwell Gas Space temperature Response (Base Case)

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Figure 3.7-9. Drywell Temperature Response (Base Case)

3.7.3 Bounding Results for Containment Analysis

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Table 3.7-1. Model Parameters for Bounding Case

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Figure 3.7-10. Containment Pressure Response (Bounding Case)

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Figure 3.7-11. PCCS Heat Removal vs. Decay Heat (Bounding Case)

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Figure 3.7-12. Suppression Pool Temperatures (Bounding Case)

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Figure 3.7-13. Drywell Pressure Response vs. Design Limit

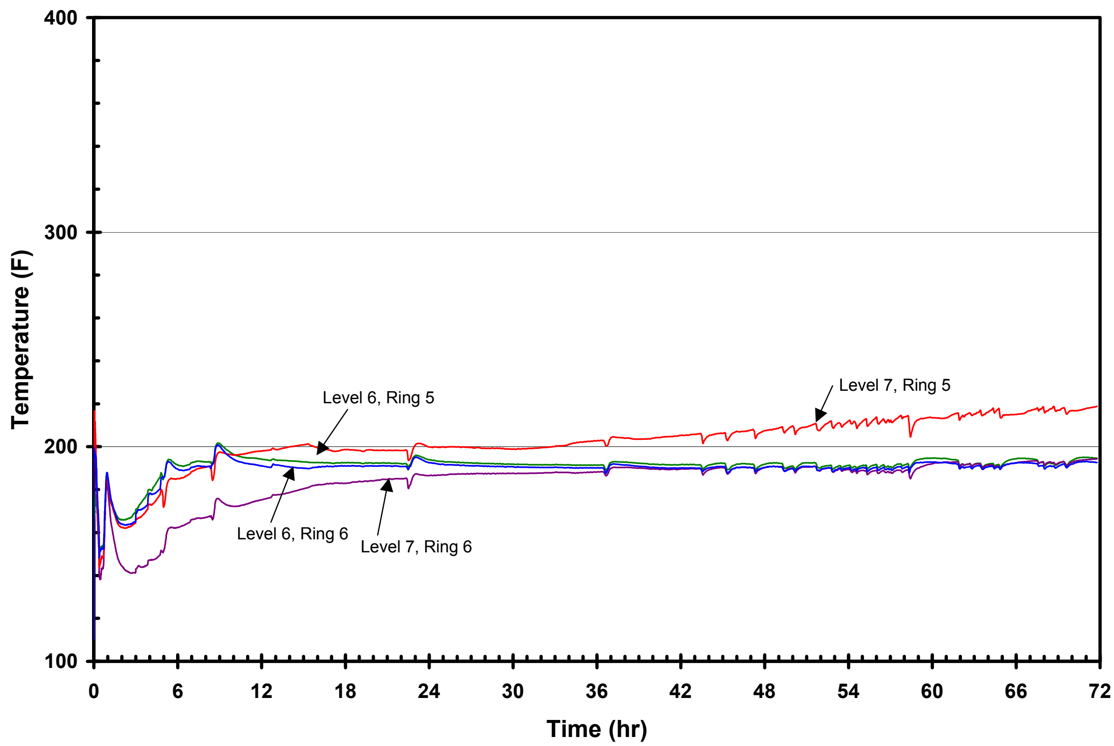

Figure 3.7-14. Wetwell Gas Space Temperature Response (Bounding Case)

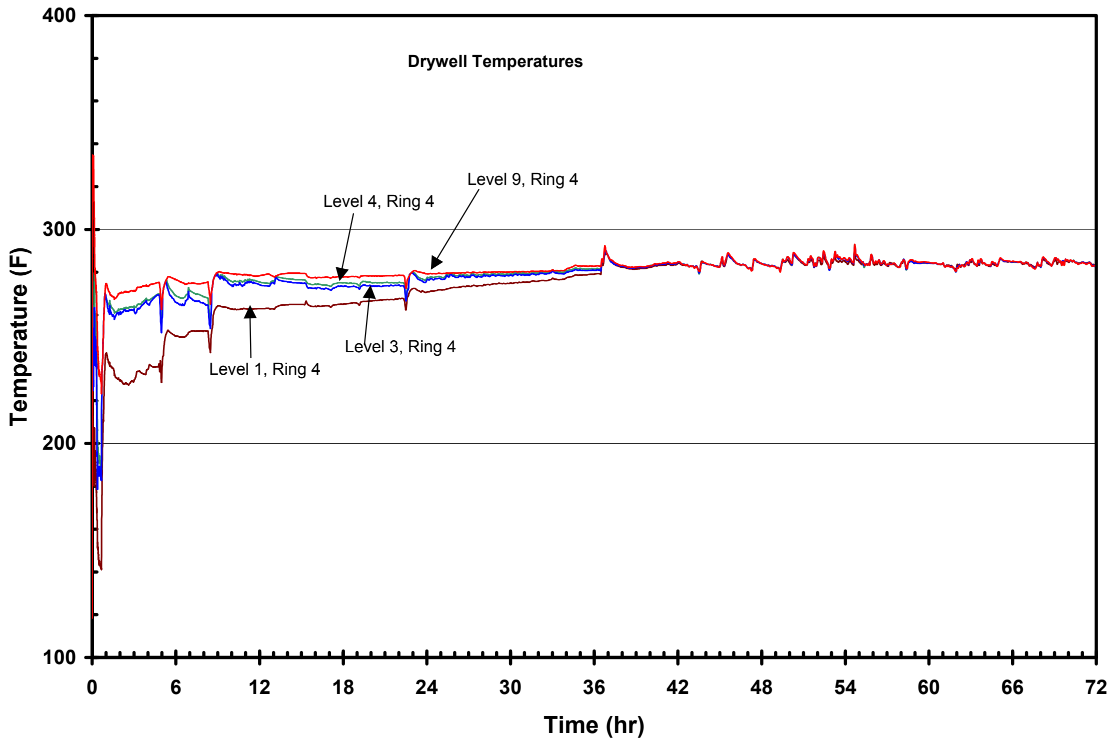

Figure 3.7-15. Drywell Temperature Response (Bounding Case)

3.8 Summary of Containment/LOCA Application Methodology

This report has defined an application methodology that meets the licensing requirements for containment /LOCA analysis for the ESBWR.

The requirements to be met and the scope of application were identified in Sections 3.1. Phenomena important for containment/LOCA analysis for ESBWR were identified in Section 3.2. Section 3.3 justified the applicability of TRACG for containment/LOCA analysis. Model and plant parameters and their ranges were established in Sections 3.4 and 3.5. A bounding application approach was proposed in Section 3.6. Results with this bounding approach were presented for the limiting main steamline break in Section 3.7 and shown to have adequate margin to the design limit.

4. TRANSIENT ANALYSIS

4.1 Licensing Requirements and Scope of Application

4.1.1 10CFR50 Appendix A

The *General Design Criteria for Nuclear Power Plants* are stipulated in Appendix A to Part 50 of 10CFR. Anticipated Operational Occurrences are classified as transient events of moderate frequency. The Standard Review Plan for events in this classification states that the "acceptance criteria are based on meeting the requirements of the following regulations" and then defines the acceptance criteria "as it relates" to the general design criteria (GDC). NRC approval of licensing methods used for AOO analysis implies that the methods are capable of assessing an AOO transient response "as it relates" to the GDC.

4.1.2 Standard Review Plan Guidelines (NUREG 800)

The NRC guidelines for review of anticipated operation occurrences (AOOs) are identified in Section 15 of the Standard Review Plan (SRP) [19].

The AOO scenarios (incidents of moderate frequency) applicable to ESBWR that can be analyzed using TRACG are listed with the corresponding SRP section.

Section	Event
15.1.1 - 15.1.4	Decrease in feedwater temperature, increase in feedwater flow, increase in steam flow, and inadvertent opening of a steam generator relief or safety valve.
15.2.1 - 15.2.5	Loss of external load; turbine trip; loss of condenser vacuum; closure of main steam isolation valve (BWR); and steam pressure regulator failure (closed).
15.2.6	Loss of non-emergency AC power to the station auxiliaries.
15.2.7	Loss of normal feedwater flow.

In addition to the events given above, there are others such as the rod withdrawal errors (Section 15.4.1.3) and fuel misloading errors (Section 15.4.7) that are analyzed with the steady-state three-dimensional core simulator PANACEA [78]. Control rod drop accidents (Section 15.4.9) are currently considered incredible events for the Fine Motion Control Rod Drives (FMCRDs) and are dispositioned generically. GE has used TRACG to perform realistic calculations for control rod drop accidents but this application is not included in the scope of the current submittal.

4.1.3 Proposed Application Methodology

The methodology for this application of TRACG to ESBWR is identical to that approved by the NRC for BWR/2-6 AOOs [3].

4.1.3.1 Conformance with CSAU Methodology

The application methodology using TRACG for ESBWR AOO transient analyses addresses all the elements of the NRC-developed CSAU evaluation methodology [15]. The CSAU report describes a rigorous process for evaluating the total model and plant parameter uncertainty for a nuclear power plant calculation. The rigorous process for applying realistic codes and quantifying the overall model and plant parameter uncertainties appears to represent the best available practice.

The CSAU methodology as documented in Reference 15 consists of 14 steps, as outlined in Table 4.1-1, which also shows where these steps are addressed for the current TRACG application.

Table 4.1-1
CODE SCALING, APPLICABILITY AND UNCERTAINTY EVALUATION METHODOLOGY

CSAU Step	Description	Addressed In
1	Scenario Specification	Section 4.1.6
2	Nuclear Power Plant Selection	Section 4.1.7
3	Phenomena Identification and Ranking	Section 4.2
4	Frozen Code Version Selection	Reference [1]
5	Code Documentation	Reference [1]
6	Determination of Code Applicability	Section 4.3
7	Establishment of Assessment Matrix	Section 4.3.2
8	Nuclear Power Plant Nodalization Definition	Section 4.4.2
9	Definition of Code and Experimental Accuracy	Reference [2],[24],[25]
10	Determination of Effect of Scale	Section 4.4.3
11	Determination of the Effect of Reactor Input Parameters and State	Section 4.5
12	Performance of Nuclear Power Plant Sensitivity Calculations	Section 4.6
13	Determination of Combined Bias and Uncertainty	Section 4.6
14	Determination of Total Uncertainty	Section 4.6

4.1.4 Implementation Requirements

The implementation of TRACG into actual licensing analysis is contingent on completion of the following implementation requirements:

- Review and approval by the NRC of:
  - The modeling uncertainties documented in Section 4.4.
  - The statistical process for analyzing AOOs described in Section **4.6**.
- ESBWR implementation using best-estimate modeling to consider sensitivities due to initial condition and plant parameters described in Sections 4.5.2 and 4.5.3.

Specific operating limits derived or comparison with acceptance criterion (peak pressure, water level, and fuel thermal/mechanical) will be based on application of the statistical application processes described in Section 4.6.

4.1.5 Review Requirements For Updates

In order to effectively manage the future viability of TRACG for AOO licensing calculations, GE proposes the following requirements for upgrades to the code to define changes that (1) require NRC review and approval and (2) that will be on a notification basis only.

4.1.5.1 Updates to TRACG Code

Modifications to the basic models described in Reference 1 may not be used for AOO licensing calculations without NRC review and approval.

Updates to the TRACG nuclear methods to ensure compatibility with the NRC-approved steady-state nuclear methods (e.g., PANAC11) may be used for AOO licensing calculations without NRC review and approval as long as the ΔCPR/ICPR, peak vessel pressure, and minimum water level shows less than 1 sigma deviation difference compared to the method presented in this LTR. A typical AOO in each of the event scenarios will be compared and the results from the comparison will be transmitted for information.

Changes in the numerical methods to improve code convergence may be used in AOO licensing calculations without NRC review and approval.

Features that support effective code input/output may be added without NRC review and approval.

4.1.5.2 Updates to TRACG Model Uncertainties

New data may become available with which the specific model uncertainties described in Section 4.4 may be reassessed. If the reassessment results in a need to change specific model uncertainty, the specific model uncertainty may be revised for AOO licensing calculations without NRC review and approval as long as the process for determining the uncertainty is unchanged.

The nuclear uncertainties (void coefficient, Doppler coefficient, and scram coefficient) may be revised without review and approval as long as the process for determining the uncertainty is unchanged. In all cases, changes made to model uncertainties done without review and approval will be transmitted for information.

4.1.5.3 Updates to TRACG Statistical Method

Revisions to the TRACG statistical method described in Section 4.6 may not be used for AOO licensing calculations without NRC review and approval.

4.1.5.4 Updates to Event Specific Uncertainties

Event specific ΔCPR/ICPR, peak pressure, and water level biases and uncertainties will be developed for AOO licensing applications based on a reference fuel type. These biases and uncertainties do not require NRC review and approval. The generic uncertainties will be transmitted to the NRC for information.

4.1.6 AOO Scenario Specification

The transient scenarios are those associated with anticipated operational occurrences (AOOs) in ESBWR. The following AOO transient events groups are specifically included:

1. Pressurization events, including: turbine trip without bypass, load rejection without bypass, feedwater controller failure increasing flow, downscale failure of pressure regulator, main steam line isolation valve closure without position scram. This grouping includes all events in SRP Section 15.2.1 - 15.2.5 that apply to BWRs. The feedwater controller failure increasing flow is in Section 15.1.1 - 15.1.4 but can also be considered a pressurization transient. The loss of auxiliary power is in SRP Section 15.2.6.
2. Depressurization events, including: upscale failure of pressure regulator. The upscale failure of pressure regulator is in SRP Section 15.1.1 - 15.1.4.
3. Cold water events, including: loss of feedwater heating. The loss of feedwater heating (decrease in feedwater temperature) is in SRP Section 15.1.1 - 15.1.4. This grouping includes all events in SRP Section 15.5.1 - 15.5.2 that apply to BWRs.
4. Level transient events such as partial or complete loss of feedwater. This grouping includes all events in SRP Section 15.2.7 that apply to BWRs.

4.1.7 Nuclear Power Plant Selection

The intended application in this report is for the ESBWR plant.

4.2 Phenomena Identification and Ranking

The critical safety parameters for AOO transients are minimum critical power ratio (MCPR), fuel thermal-mechanical margins, downcomer water level and peak reactor pressure vessel (RPV) pressure. These are the criteria used to judge the performance of the safety systems and the margins in the design. The values of the critical safety parameters are determined by the governing physical phenomena. To delineate the important physical phenomena, it has become customary to develop phenomena identification and ranking tables (PIRTs). PIRTs are ranked with respect to their impact on the critical safety parameters. For example, the MCPR is determined by the reactor short-term response to transients. The coupled core neutronic and thermal-hydraulic characteristics govern the neutron flux, reactor pressure, core flow and downcomer water level transients.

Section 4.2.1 describes representative scenarios for ESBWR AOOs. The descriptions stress the phenomenological evolution of the transients. The scenarios provide a background for the listing and ranking of phenomena that go into the PIRT. Section 4.2.2 reports the results of the phenomena ranking from Reference 29.

4.2.1 ESBWR AOO Classes

The PIRTs for anticipated transients were synthesized from consideration of the phenomena involved in various classes of events.

4.2.1.1 Fast Pressurization Events

These are the limiting pressurization events. Principal figures of merit on which “importance” is defined are critical power (MCPR) and reactor pressure.

- ***Turbine Trips*** — initiated by trip of turbine stop valves from full open to full closed. Analyzed with bypass valves functional, and with bypass failure.

- ***Generator Load Rejection*** — initiated by fast closure of turbine control valves from partially open position to full-closed. This event is analyzed with bypass valves functioning, and with bypass failure. The turbine control valves may be initially at the same position (full arc turbine admission) or at different positions (partial arc turbine admission).

- ***Loss of AC Power*** — Similar to load rejection; however, bypass valves are assumed to close after 6 seconds due to loss of power to condenser circulating water pumps.

- ***Main Steamline Isolation Valve (MSIV) Closure*** — In this case, the scram signal on valve position is further in advance of complete valve closure. This effectively mitigates the shorter line length to the vessel available as a compression volume.

- ***Loss of Condenser Vacuum*** — This event is similar to the Loss of AC Power and a Turbine Trip with Bypass. Because a turbine trip occurs at a higher vacuum setpoint than the bypass valve isolation, the bypass valves are available to mitigate the initial pressure increase.

4.2.1.2 Slow Pressurization Events

These are analyzed principally to ensure that they are bounded by the fast pressurization events. MCPR and reactor pressure determine "importance."

- ***Pressure Regulator Downscale Failure*** — Simultaneous closure of all turbine control valves in normal stroke mode. The triplicated fault tolerant control system prevents any single failure from causing this and makes its frequency below the anticipated abnormal occurrence category.

- ***Single Control Valve Closure*** — This event could be caused by a hydraulic failure in the valve or a failure of the valves rotor/actuator.

4.2.1.3 Decrease in Reactor Coolant Inventory

Loss of feedwater flow is characteristic of this category of transient. The IC maintains downcomer water level. Reactor water level in the downcomer is the principal figure of merit on which "importance" is defined.

4.2.1.4 Decrease in Moderator Temperature

These events challenge MCPR and stability, which are the figures of merit on which "importance" is defined:

- Loss of Feedwater Heating — initiated by isolation or bypass of a feedwater heater.

- Feedwater Controller Failure — hypothesizes an increase in feedwater flow to the maximum possible with all four feed pumps operating at maximum speed. This event is similar to turbine trip but with more severe power transient due to colder feedwater.

To determine the phenomena important in modeling anticipated transients, the sequence of events and system behavior for each class of events should be understood. To provide an example of this, the sequence of events for a fast pressurization transient is discussed below. For this class of transients, important phenomena are those affecting the MCPR and reactor pressure.

4.2.1.5 Generator Load Rejection Event Description

A fast pressurization event will occur due to the fast closure of the turbine control valves (TCVs), which can be initiated when electrical grid disturbances occur which result in significant loss of electrical load on the generator. Closure of the turbine stop valves is initiated by the turbine protection system. The valves are required to close rapidly to prevent excessive overspeed of the turbine-generator rotor.

At the same time, the turbine stop or control valves are signaled to close, and the turbine bypass valves are signaled to open in the fast opening mode. The bypass valves are fully open only slightly later than the turbine valves are closed, and can relieve more than one-third of rated steam flow to the condenser, greatly mitigating the transient. The bypass valves also use a triplicated digital controller. No single failure can cause all turbine bypass valves to fail to open on demand. The worst single failure can only cause one turbine bypass valve to fail to open on demand.

The closing time of the TCVs is short relative to the sonic transit time of the steamline, so their closure sets up a pressure wave in the steamlines. When the pressure wave reaches the vessel steam dome, the flow rate leaving the vessel effectively undergoes a step change. The area change entering the steam dome partially attenuates the pressure wave, propagating a weaker pressure disturbance down through the chimney and downcomer, increasing the vessel pressure, and reducing voids in the core. The void-reactivity feedback results in an increase in the neutron flux. A reflection of the pressure wave also travels back toward the turbine, producing an oscillation in flow and pressure in the steamlines.

Concurrent with closure of the turbine control valves, a scram condition is sensed by the reactor protection system. A turbine stop valve position less than approximately full open triggers a scram, as does the low hydraulic fluid pressure in the turbine control valve solenoids that start their fast closure mode. The ESBWR digital multiplexed Safety System Logic Control (SSLC) will initiate a scram when any two turbine stop valves are sensed as closing, or any two turbine control valves are sensed as fast closing.

The core reactivity is decreased by the control blade insertion and increased by the decrease in core voids and increase in inlet flow. The net effect may be either an immediate shutdown of the reactor and decrease in neutron flux (in cases where there are control blades partially inserted in high worth areas of the core) or a short period of increased reactivity and neutron flux followed by shutdown (in the safety analysis case where there are no control blades initially inserted, and a slower bounding CRD scram insertion time is assumed.)

In the case where the neutron flux undergoes a transient increase, the energy deposition in the fuel pellet will increase clad heat flux. The minimum value of critical power ratio during this transient is found to occur in the upper part of the bundle.

Eventually, as the blades are fully inserted, the reactor is driven subcritical, power drops to decay heat levels, and clad temperature equilibrates near saturation temperature.

The vessel pressure increase is terminated by the bypass valve opening. The downcomer water level drops below the feedwater sparger and sprays subcooled water into the steam dome. This quenching of vapor also helps to terminate the pressure increase. If the bypass and feedwater systems are assumed to be unavailable, the duration of increased pressure would be long enough to initiate the isolation condenser.

In the ASME overpressure protection analysis, the Isolation Condenser is not considered, causing the pressure to slowly increase to the SRV opening pressure. The pressure increase is terminated immediately with SRV activation, and the maximum vessel pressure occurs at the vessel bottom. The overpressure protection case conservatively assumes the first scram signal to fail, and scram on neutron flux terminates the power increase in both turbine valve closure and the MSIV closure events.

The downcomer water level response in pressurization events is driven by the transfer of water from the downcomer to core and chimney caused by the collapse of voids in the core and chimney regions. The sensed water level decreases rapidly below the L3 low water scram setpoint. The feedwater system flow increases fast enough to prevent the L2 setpoint being reached in high frequency events (events where feedwater and bypass valves are available). The feedwater control system will demand maximum feedwater flow for approximately one minute, until normal downcomer water level is restored. Without feedwater, the downcomer level drop will progress to L2, initiating the IC, isolating the MSIVs and transferring the CRD system to high-pressure injection mode. The IC can independently maintain the downcomer water level near the L2 setpoint. CRD high-pressure injection will cause the downcomer water level to slowly recover to above normal, and then automatically trip off.

4.2.2 Phenomena Identification and Ranking Table (PIRT) for AOOs

A table was developed to identify the phenomena that govern ESBWR AOO transient responses in Reference 29. 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 ranking of the phenomena is done on a scale of high importance to low importance or not applicable, as defined by the following categories:

- *High importance (H):* These phenomena have a significant impact on the primary safety parameters and should be included in the overall uncertainty evaluation. The table for High ranked phenomena has been extracted from Reference 29 and is shown in Table 4.2-1. An example of such a parameter would be the *void coefficient* for a pressurization event (C1AX in Table 4.2-1). The void coefficient determines the amount of reactivity change due to void collapse during the transient.

- *Medium importance (M):* These phenomena have insignificant impact on the primary safety parameters and may be excluded in the overall uncertainty evaluation. An example of such a parameter would be *flashing in the core* for a depressurization event. Vapor production due to fuel heat transfer dominates the effect of flashing in the core.

- *Low importance (L) or not applicable (N/A):* These phenomena have no impact on the primary safety parameters and need not be considered in the overall uncertainty evaluation. An example of such phenomenon would be *lower plenum stratification* during a pressurization event. The pressurization event happens so quickly that even if there were significant thermal stratification in the lower plenum, it could not impact the critical parameters before the event was over.

The PIRT serves a number of purposes. First, the phenomena are identified and compared to the modeling capability of the code to assess whether the code has the necessary models to simulate the phenomena. Second, the identified phenomena are cross-referenced to the qualification basis to determine what qualification data are available to assess and qualify the code models and to determine whether additional qualification is needed for some phenomena. As part of this assessment, the range of the PIRT phenomena covered in the tests is compared with the corresponding range for the intended application to establish that the code has been qualified for the highly ranked phenomena over the appropriate range.

Finally, uncertainties in the modeling of the highly ranked PIRT phenomena are carefully evaluated, and then combined through a statistical process, to arrive at the total model uncertainty. In this third stage, one may find that some highly ranked phenomena do not contribute significantly to the overall uncertainty even when conservative values for the individual phenomena uncertainties are used. It is at this stage that one can determine how individual uncertainties influence the total uncertainty so that the effort can be focused on establishing the uncertainties for those phenomena that have the greatest impact on the critical safety parameters. These uncertainties will be more fully developed later in this report.

Table 4.2-1 Composite List of Highly Ranked Phenomena for ESBWR Transients

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4.2-1 Composite List of Highly Ranked Phenomena for ESBWR Transients (Continued)

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4.3 Applicability of TRACG to Transient Analysis

The objective of this section is to demonstrate the applicability of TRACG for the analysis of anticipated transient events in ESBWR. To accomplish this purpose, the capability of the TRACG models to treat the highly ranked phenomena and the qualification assessment of the TRACG code for AOO applications is examined in the next two subsections.

4.3.1 Model Capability

The capability to calculate an event for a nuclear power plant depends on four elements:

- Conservation equations, which provide the code capability to address global processes.
- Correlations and models, which provide code capability to model and scale particular processes.
- Numerics, which provide code capability to perform efficient and reliable calculations.
- Structure and nodalization, which address code capability to model plant geometry and perform efficient and accurate plant calculations.

Consequently, these four elements must be considered when evaluating the applicability of the code to the event of interest for the nuclear power plant calculation. The key phenomena for each event are identified in generating the PIRTs for the intended application, as indicated in Section 4.2.2. The capability of the code to simulate these key phenomena is specifically addressed, documented, and supported by qualification in References 2 and 24.

Important ESBWR phenomena have been identified and TRACG models have been developed to address these phenomena as indicated in Table 4.3-1 for the high ranked phenomena. The models are identified so that they may be easily correlated to the model description sections.

Table 4.3-1

High Ranked ESBWR Phenomena and TRACG Model Capability Matrix

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4.3.2 Model Assessment Matrix

The qualification of TRACG models is summarized in Table 4.3-2. For each of the governing ESBWR phenomena, TRACG qualification has been performed against a wide range of data. In this section, the qualification basis is related to the phenomena that are important for the intended application. This is a necessary step to confirm that the code has been adequately qualified for the intended application.

The list of High ranked phenomena is cross-referenced to the qualification basis in Table 4.3-2. Data from separate effects tests, component tests, integral system tests and plant tests as well as BWR plant data have been used to qualify the capability of TRACG to model the phenomena.

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Table 4.3-2
QUALIFICATION MATRIX FOR HIGH RANKED PHENOMENA FOR ESBWR TRANSIENTS

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4.4 Model Uncertainties and Biases

Overall model biases and uncertainties for a particular application are assessed for each high ranked phenomena by using a combination of comparisons of calculated results to: (1) separate effects test facility data, (2) integral test facility test data, (3) component qualification test data and (4) BWR plant data. Where data is not available, cross-code comparisons or engineering judgment are used to obtain approximations for the biases and uncertainties. Some medium ranked phenomena have also been included where it was felt the effects were not negligible. For some phenomena that have little impact on the calculated results, it is appropriate to simply use a nominal value or to conservatively estimate the bias and uncertainty.

The phenomena for ESBWR AOO transients have already been identified and ranked, as indicated in Section 4.2. For the high ranked phenomena, the bases used to establish the nominal value, bias and uncertainty for that parameter are documented in Section 4.4.1. Also, the basis for the selection of the probability density function used to model the uncertainty is provided in Section 4.4.1. The bias and uncertainty are implemented in TRACG through special input parameters designated as "PIRT multipliers".

4.4.1 Model Parameters and Uncertainties

This section discusses the uncertainties associated with each item from Table 4.2-1 (list of highly ranked parameters). Some medium ranked parameters have also been included. The results are summarized in Table 4.4-1.

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Figure 4.4-1. Void Coefficient Normalized %Bias and %Standard Deviation [3]

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Table 4.4-1

High Ranked Model Parameters for AOO analysis

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4.5 Application Uncertainties and Biases

4.5.1 Input

Specific inputs for each transient event are specified via internal procedures, which are the primary means used by GE to control application of engineering computer programs. The specific code input will be developed in connection with the application LTR and the development of the application specific procedure. This section will be limited to a more general discussion of how input is treated with respect to quantifying their impact on the calculated results. As such, it serves as a basis for the development of the application specific procedures.

Code inputs can be divided into four broad categories: (1) geometry inputs; (2) model selection inputs; (3) initial condition inputs; and (4) plant parameters. For each type of input, it is necessary to specify the value for the input. If the calculated result is sensitive to the input value, then it is also necessary to quantify the uncertainty in the input.

The geometry inputs are used to specify lengths, areas and volumes. Uncertainties in these quantities are due to measurement uncertainties and manufacturing tolerances. These uncertainties usually have a much smaller impact on the results than do other uncertainties associated with the modeling simplifications. When this is not the case, the specific uncertainties can usually be quantified in a straightforward manner. For example, consider the 2% channel flow area uncertainty that is considered as part of the Safety Limit MCPR (SLMCPR). This uncertainty is determined from the manufacturing tolerances on the inner dimensions of the channel box and the outer diameter of the fuel and water rods. It is known that neglecting this uncertainty causes the calculated SLMCPR value to be non-conservative by no more than 0.0015. Even though channel flow area is considered to be *important*, the impact associated with the uncertainty in this parameter is small.

Individual geometric inputs are the building blocks from which the spatial nodalization is built. Another aspect of the spatial nodalization includes modeling simplifications such as the lumping together of individual elements into a single model component. For example, several similar fuel channels may be lumped together and simulated as one fuel channel group. An assessment of these kinds of simplifications, along with the sensitivities to spatial nodalization, is included in the *TRACG Qualification* [2].

Model selection inputs are used to select the features of the model that apply for the intended application. Once established, these inputs are fully specified in the procedure for the application and will not be changed.

A distinction has been made in this document between *initial conditions* and *plant parameters*. Obviously, when specified in absolute units, the initial rated conditions for a nuclear power plant are specific to the plant and thus have in some documents been considered as plant parameters. In this document we consider *initial conditions* to be those key plant inputs that determine the overall steady-state nuclear and hydraulic conditions prior to the transient. These are inputs that are essential to determining that the steady-state condition of the plant has been established. Initial conditions parameters and the uncertainties associated with them are addressed in Section 4.5.2.

The name *plant parameter*, on the other hand, is reserved for such things as protection system setpoints, valve capacities and stroke times, and scram characteristics that influence the characteristics of the transient response but which do not (when properly prescribed) have an impact on steady-state operation. Plant parameters and the uncertainties associated with them are addressed in Section 4.5.3.

4.5.2 Initial Conditions

Initial conditions are those conditions that define a steady-state operating condition. Initial conditions for a particular transient scenario are specified in the procedure for the application. For example, the procedure may specify that the calculation be performed at the end-of-cycle exposure at 100% of rated power and flow using a power and exposure distribution that has been obtained from a prescribed process.

Initial conditions may vary due to the allowable operating range or due to uncertainty in the measurement at a give operating condition. The plant Technical Specifications and Operating Procedures provide the means by which controls are instituted and the allowable initial conditions are defined. At a given operating condition, the plant's measurement system has inaccuracies that also must be accounted for as an uncertainty. The key plant initial conditions are identified in Table 4.5-1.

The analyses performed must maintain consistency with the allowed domains of operation. The impact of the initial condition on the results are characterized in the following manner:

- The results are sensitive to the initial condition and a basis for the limiting initial condition cannot be established. Future plant analyses (e.g., the reload licensing analyses) will consider the full allowable range of the initial condition.

- The results are sensitive to the initial condition and a basis for the limiting initial condition can be established. Future plant analyses (e.g., the reload licensing analyses) will consider the parameter to be at its limiting initial condition.

- The results are not sensitive to the initial condition and a nominal initial condition will be assumed for the parameter.

Each initial condition is monitored through the use of plant sensors or simulated prediction. Because of instrument or simulation uncertainty, the plant condition may vary from the indicated value. The results are characterized in the following manner:

- The results are sensitive to the uncertainty in the initial condition and the uncertainty in the initial condition will be included in the statistical analysis.

- The results are not sensitive to the uncertainty in the initial condition and the uncertainty does not need to be accounted for.

The impact of the total uncertainty in initial conditions must also be quantified for the critical safety parameters such as ΔCPR/ICPR, peak vessel pressure and water level. Some of these uncertainties may already be considered by other means. The evaluation, which addresses the characterization, is contained in this section.

Table 4.5-1

KEY PLANT INITIAL CONDITIONS

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4.5.3 Plant Parameters

A *plant parameter* is defined as a plant-specific quantity such as a protection system setpoint, valve capacity or stroke time, or a scram characteristic, etc. *Plant parameters* influence the characteristics of the transient response and have essentially no impact on steady-state operation, whereas *initial conditions* are what define a steady-state operating condition.

For each plant parameter, a conservative value corresponding to the *analytic limit* is defined. The analytic limit (AL) is the value used for the transient licensing analyses. In many cases, the value used for the AL can be related to a plant Technical Specifications, since most of the plant parameter values that are important for AOO transient responses are related to processes that are controlled by the plant Technical Specifications. These parameters may be periodically measured at the plants to assure compliance with the Technical Specifications. Performance and uncertainties for the processes that the Technical Specifications are designed to control are based on manufacturing specifications, performance data, as well as required surveillance. A Technical Specification value will usually be in terms of a maximum or minimum acceptable value that bounds the entire population of values that are measured at the plant.

The Technical Specifications values may be used to define the analytic limits used for the licensing analyses. The original licensing basis specified bounding Technical Specifications values for most of the plant parameters. This is one acceptable way by which conservatism can be added to a "best estimate" methodology. Another option for establishing plant parameters is to establish an uncertainty in the parameter. For example, the NRC has accepted (AOO analysis *Option B* for operating plants) a faster scram speed when used together with considerations of the uncertainties in the scram speeds. This approach is supported by surveillance procedures at the plant, whereby the scram times are measured. The uncertainty in the scram times is then accounted for in the AOO analyses as part of the statistical methodology.

GE procedures will define the critical Operating Parameters for Licensing (OPL) for transient analysis. It serves as a guide for generating plant parameter data to be used for licensing. This procedure addresses Technical Specifications items as well as other items that are important to the severity of transients.

The reactor scram is the most effective plant system for mitigating the severity of a transient. The plant Technical Specifications provide surveillance requirements to ensure control rod operability and scram times. The scram times used for the analysis depend on the type of transient analyzed. Table 4.5-2 shows the analytical scram speed characteristics for the ESBWR. These are based on the ABWR. Because the control rod stroke is shorter, rod motion is slower than for ABWR.

Table 4.5-2
ANALYTICAL SCRAM SPEEDS FOR ESBWR

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4.6 Combination of Uncertainties

A proven Monte Carlo technique is used to combine the individual biases and uncertainties into an overall bias and uncertainty. The Monte Carlo sample is developed by performing random perturbations of model and plant parameters over their individual uncertainty ranges. Using the histogram generated by the Monte Carlo sampling technique, a probability density function is generated for code output of the primary safety criteria parameters.

In order to determine the total uncertainty in predictions with a computer code, it is necessary to combine the effects of model uncertainties (CSAU Step 9), scaling uncertainties (CSAU step 10), and plant condition or state uncertainties (CSAU Step 11). Various methods have been used to combine the effects of uncertainties in safety analysis. This section summarizes the method used for combining uncertainties for the AOO application. This is the same approach that has been successfully used and approved for analyses of AOO transients for operating plants [3].

4.6.1 Recommended Approach for Combining Uncertainties

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4.6.1.1 Order Statistics (OS) Method – Single Bounding Value

The Monte Carlo method that has been used in Germany by Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) [83] requires only a modest number of calculations, and automatically includes the effects of interactions between perturbations to different parameters. In the OS method, Monte Carlo trials are used to vary all uncertain model and plant parameters randomly and simultaneously, each according to its uncertainty and assumed probability density function (PDF), and then a method based on the order statistics of the output values is used to derive upper tolerance bounds (one-sided, upper tolerance limits OSUTLs).

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An OSUTL is a function $U = U(x_1,\ldots,x_n)$ of the data $x_1,\ldots,x_n$ (which will be the values of an output parameter of interest in a set of Monte Carlo trials), defined by two numbers
$0 < \alpha$, $\beta < 1$, so that the proportion of future values of the quantity of interest that will be less than U is 100α%, with confidence at least 100β% --- this is called an OSUTL with 100α%-content and (at least) 100β% confidence level.

The order statistics method, originally developed by Samuel Wilks, produces OSUTLs that are valid irrespective of the probability distribution of the data, requiring only that they be a sample from a continuous PDF. Given values of α and β, the OSUTL can be defined as the largest of the data values, provided the sample size $n \geq \log(1-\beta)/\log\alpha$ [84]. For 95%-content and 95% confidence level, the minimum sufficient sample size is n=59.

The order statistics method is generally applicable, irrespective of the probability distribution of the data, and requires only that these be like outcomes of independent random variables with a common probability distribution.

If the method is implemented as described above, whereby the sample size (59) was chosen so that the sample maximum is the upper tolerance bound sought (95% content with 95% confidence), then this bound, as a random quantity, has variability that is typical of the maximum of a sample of that size, which can be substantial, and occasionally may yield an overly conservative bound.

To mitigate this variability, one can choose a suitably larger sample size so that the bound sought is now given by the second or third largest sample value. For example, the 95% content with 95% confidence tolerance bound is the third largest observation in a sample of size 124: Just for the sake of illustration, in normal (that is, Gaussian) populations its variability is about one half of the variability of the maximum in a sample of size 59; and in the more heavily-tailed Student's t distribution with 4 degrees of freedom, the variability of the third largest in a sample of size 124 is about one third of the variability of the maximum in a sample of size 59.

The following table summarizes the sample sizes that are required, when the bound is the largest, the second largest, or the third largest order statistic, all for 95% content and 95% confidence:

Order Statistic	Sample Size
Largest	59
2nd Largest	93
3rd Largest	124

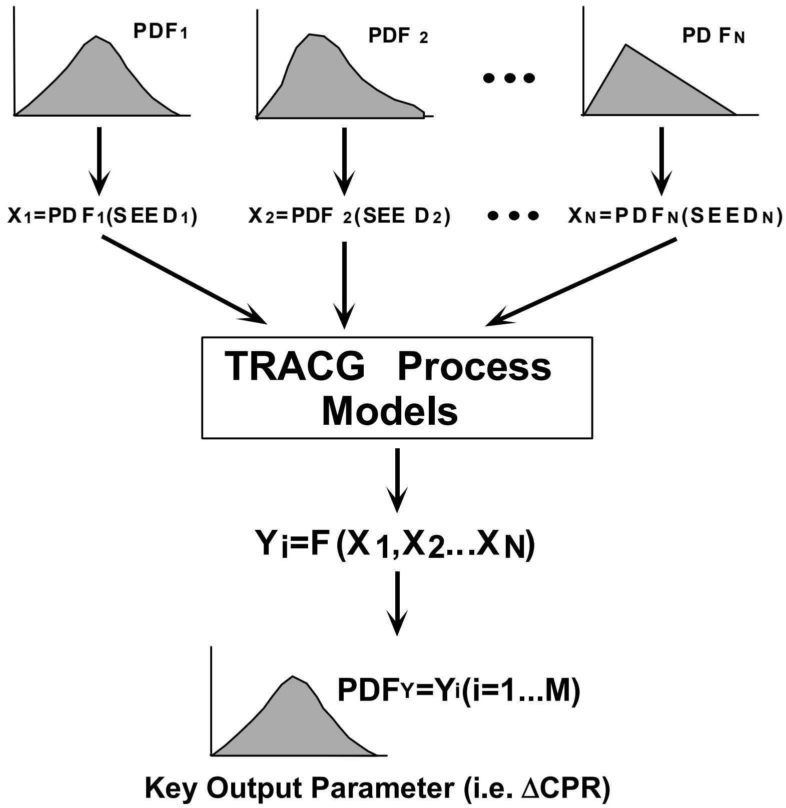

Figure 4.6-1. Schematic Process for Combining Uncertainties

4.6.1.2 Normal Distribution One-Sided Upper Tolerance Limit

If the data that the tolerance bound will be derived from can reasonably be regarded as a sample from a normal (that is, Gaussian) probability distribution, then this normal distribution one-sided upper tolerance limit (ND-OSUTL) is of the form

$$ND - OSUTL_{\alpha,\beta} \equiv \bar{y} + z_{\alpha,\beta} \cdot s$$

where $\bar{y}$ denotes the average of the outcomes of the TRACG trials, and s denotes their standard deviation, and the factor $z_{\alpha,\beta}$ is chosen to guarantee 100α%-content and 100β% confidence level. Since this factor $z_{\alpha,\beta}$ depends on the assumption of normality for the data, one must first ascertain whether the data does indeed conform with the Gaussian model, typically using one or several goodness-of-fit tests: for example, Ryan-Joiner's, Shapiro-Wilk's, or Anderson-Darling's. The values of $z_{\alpha,\beta}$ are tabulated in many statistical textbooks [86] as *factors for one-sided normal tolerance limits*. For example, for a sample of size $n = 59$, and a 95% content and a 95% confidence level, $z_{95,95}$ = 2.024. As the sample size n increases, this factor approaches 1.645, the 95th percentile of the standard normal distribution. Unlike the order statistics method, this ND-OSUTL method does not require specific minimum sample sizes; but it does require normality. If the data are unlikely to have originated from a normal population, then one should use the order statistics method.

4.6.1.3 Advantages of Recommended Method

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4.6.2 Implementation of Statistical Methodology

The purpose of this section is (1) to describe the process by which the statistical results will be used to determine the Operating Limit Minimum Critical Power Ratio (OLMCPR), and (2) establish that fuel thermal/mechanical performance, peak vessel pressure, and minimum water level have acceptable margins to design limits. The application to the latter three is straightforward, and is discussed in the next section. The determination of the OLMCPR is more involved, and is detailed in the subsequent sections.

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Figure 4.6-2. Generic ΔCPR/ICPR Uncertainty Development

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Figure 4.6-3. NRSBT Determination

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Figure 4.6-4. GESAM Calculation Procedure for Analytical Determination of OLMCPR

4.7 Demonstration Calculations for ESBWR AOOs

The analyses provided in this Section form the bases for future application of TRACG to ESBWR AOOs. TRACG performance is demonstrated on one or more limiting licensing basis events for the scenarios specified in Section 4.2.1. This demonstration includes baseline TRACG analysis for a representative core. Statistical calculations for the various limiting AOOs will be performed for the ESBWR for the final core design utilizing the process described in Section 4.6.

4.7.1 Baseline Analysis [[

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Table 4.7-1. Sequence of Events for LRNB Transient

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Table 4.7-2. Sequence of Events for FWCF Event

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Table 4.7-3. Sequence of Events for MSIV Closure Transient

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Figure 4.7-1. Pressure response for LRNB Transient

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Figure 4.7-2. Neutron Flux Response for LRNB Transient

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Figure 4.7-3. Downcomer Two-Phase Level Response for LRNB Transient

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Figure 4.7-4. Bundle Power Response for LRNB Transient

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Figure 4.7-5. Bundle Inlet Flow for LRNB Transient

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Figure 4.7-6. Downcomer Level Response for FWCF Transient

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Figure 4.7-7. Pressure Response for FWCF Transient

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Figure 4.7-8. Neutron Flux response for FWCF Transient

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Figure 4.7-9. Bundle Power Response for FWCF Transient

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Figure 4.7-10. Bundle Inlet Flow Response for FWCF Transient

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Figure 4.7-11. Pressure Response for MSIV Closure Transient

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Figure 4.7-12. Neutron Flux Response for MSIV Closure Transient

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Figure 4.7-13. Downcomer Level for MSIV Closure Transient

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Figure 4.7-14. IC Steam Flow for MSIV Closure Transient

4.8 Summary of TRACG Application to ESBWR AOOs

This report has provided the basis for extending the application methodology that has been approved for operating BWRs for AOOs to the ESBWR.

The requirements to be met and the scope of application were identified in Section 4.1. Phenomena important for AOO analysis for ESBWR were identified in Section 4.2. Section 4.3 justified the applicability of TRACG for ESBWR AOO analysis. Model and plant parameters and their ranges were established in Sections 4.4 and 4.5. A statistical application approach (identical to that approved for operating plants) was proposed in Section 4.6. Sample base line analyses were shown in Section 4.7 for three different pressurization transients to illustrate ESBWR response and demonstrate that it is generally similar to operating plants.

Actual application to the ESBWR SAR calculations will involve repeating the baseline analysis for the final ESBWR core; performing sensitivity studies for model and plant parameters; and performing a statistical analysis in conformance with the process described in Section 4.6.

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NEDO-33083-A Revision 1

Attachment 1

NRC SAFETY EVALUATION

ADDENDUM TO THE SAFETY EVALUATION REPORT WITH OPEN ITEMS FOR NEDC-33083P-A, "APPLICATION OF THE TRACG COMPUTER CODE TO THE ECCS AND CONTAINMENT LOCA ANALYSIS FOR THE ESBWR DESIGN"

September 20, 2010

Mr. Jerald G. Head
Senior Vice President, Regulatory Affairs
GE Hitachi Nuclear Energy
3901 Castle Hayne Road MC A-18
Wilmington, NC 28401

SUBJECT: FINAL SAFETY EVALUATION FOR GE HITACHI NUCLEAR ENERGY ADDENDUM TO THE SAFETY EVALUATION FOR LICENSING TOPICAL REPORT NEDC-33083P-A, "APPLICATION OF THE TRACG COMPUTER CODE TO THE ECCS AND CONTAINMENT LOCA ANALYSIS FOR ESBWR DESIGN"

Dear Mr. Head:

On August 24, 2005, GE Hitachi (GEH) Nuclear Energy submitted the Economic Simplified Boiling Water Reactor (ESBWR) design certification application to the staff of the U.S. Nuclear Regulatory Commission. Subsequently, in support of the design certification, GEH submitted the addendum to the safety evaluation for license topical report (LTR) NEDC-33083P-A, "Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design." The staff has now completed its review of NEDC-33083P-A.

The staff finds NEDC-33083P-A, "Application of the TRACG Computer Code to the ECCS and Containment LOCA Analysis for the ESBWR Design," acceptable for referencing for the ESBWR design certification to the extent specified and under the limitations delineated in the LTR and in the associated safety evaluation (SE). The SE, which is enclosed, defines the basis for acceptance of the LTR.

The staff requests that GEH publish the revised proprietary and non-proprietary versions of the LTR listed above within 1 month of receipt of this letter. The accepted version of the topical report shall incorporate this letter and the enclosed SE and add an "-A" (designated accepted) following the report identification number.

If NRC's criteria or regulations change, so that its conclusion that the LTR is acceptable is invalidated, GEH and/or the applicant referencing the LTR will be expected to revise and resubmit its respective documentation, or submit justification for continued applicability of the LTR without revision of the respective documentation.

Document transmitted herewith contains sensitive unclassified information. When separated from the enclosures, this document is "DECONTROLLED."

Pursuant to 10 CFR 2.390, we have determined that the enclosed SE contains proprietary information. We will delay placing the non-proprietary version of this document in the public document room for a period of 10 working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects only. If you believe that any additional information in Enclosure 1 is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.390.

The Advisory Committee on Reactor Safeguards (ACRS) subcommittee, having reviewed the subject LTR and supporting documentation, agreed with the staff's recommendation for approval following the August 16, 2010 ACRS subcommittee meeting.

Sincerely,

/RA Frank Akstulewicz for:/

David B. Matthews, Director
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-010

Enclosure:
1. Safety Evaluation (Non-Proprietary)
2. Safety Evaluation (Proprietary): Applicant only

cc: See next page

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cc: See next page

ADAMS ACCESSION NO. - ML102500164-Package

OFFICE	BWR:PM	BWR:LA	BC:SRSB	BWR:LPM	BWR:BC	OGC/NLO	DNRL:D
NAME	BBavol	SGreen	JDonoghue	ACubbage	MTonacci	MCarpentier	DMatthews
DATE	09/08/10	09/14/10	09/17/10	09/08/10	09/20/10	09/08/10	09/20/10

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SUBJECT: FINAL SAFETY EVALUATION FOR GE HITACHI NUCLEAR ENERGY ADDENDUM TO THE SAFETY EVALUATION FOR LICENSING TOPICAL REPORT NEDC-33083P-A, "APPLICATION OF THE TRACG COMPUTER CODE TO THE ECCS AND CONTAINMENT LOCA ANALYSIS FOR ESBWR DESIGN" DATED SEPTEMBER 20, 2010

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

GE Hitachi Nuclear Energy, LLC (GEH) submitted topical report NEDC-33083P, "TRACG Application for ESBWR," in November 2002, during the preapplication phase of the economic simplified boiling-water reactor (ESBWR) design certification review. The staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed and accepted the GEH TRACG code for analyzing loss-of-coolant accident (LOCA) events for the ESBWR design with confirmatory items (Reference 1)[1]. In addition, from December 11 through December 15, and resuming for the period between December 19 and December 20, 2006, the NRC staff conducted an audit of the TRACG code as it is applied to ESBWR LOCA analyses to evaluate updates to the code and methodology since its original approval (Reference 2). The detailed basis for the staff's approval of TRACG is described in the Safety Evaluation Report, which is incorporated in the proprietary approved version, NEDC-33083P-A (Reference 1). Hereafter, all citations to Reference 1 apply to the staff safety evaluation, unless noted otherwise.

The staff documented "confirmatory items" during this review. The staff stated that these items "were identified as needing confirmation at the design certification stage. These items do not affect the applicability or the capability of the code, but do address the response of the plant design and adequacy of the documentation." The Summary of TRACG LOCA SER Confirmatory Items (Summary of September 9, 2005 NRC/GE Conference Call on TRACG LOCA SER Confirmatory Items) (Reference 3) identifies the confirmatory items and the planned GEH actions for each item. In Reference 4, GEH provided Design Certification information for Confirmatory Item 1 related to the Reactor Pressure Vessel (RPV) Level Response for the Long Term PCCS Period, and identified major design changes from the pre-Application review to the Design Certification Document (DCD) design. Reference 5 provides information requested by the staff in the Acceptance Review for NEDC-33083P. Reference 6 is a revised response to the Acceptance Review items which incorporates changes to the TRACG model representing the feedwater line break.

The following safety evaluation report (SER) addendum documents the staff's evaluation of these items. Each section contains the confirmatory item directly quoted from Section 4.0 of the staff SER (the approved staff safety evaluation for TRACG application for ESBWR) (Reference 1).

1 Item 1: Phenomena Identification and Ranking Table for Long-Term Core Cooling

1.1 <u>Confirmatory Item 1</u>

"The PIRT at the design certification stage should include the long-term cooling phase of the LOCA since the long-term cooling phase is highly design dependent. Should it be found that unreviewed phenomena occur during the long-term cooling phase, the appropriate models and correlations in the TRACG code will be revisited by the staff."

[1] See ADAMS Accession No. ML051390265 pages 11 through 185.

1.2 Staff Evaluation of Confirmatory Item 1

In support of the design certification application, and to satisfy pre-application confirmatory items, GEH submitted details on long-term core cooling in Reference 4 and in Chapter 6, Section 6G, of the ESBWR DCD Revision 5 (Reference 34). GEH included a discussion of long-term inventory distribution for four break locations—(1) main steamline break (MSLB), (2) feedwater line break (FWLB), (3) bottom drainline break (BDLB), and (4) gravity-driven cooling system (GDCS) line break (GDLB).

The requirements for a realistic methodology in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46 are somewhat different than those for a prescriptive methodology in that more realistic models can be used and a measure of the uncertainty in the code must be determined. Various means of achieving an estimate of uncertainty are available. GEH has chosen to follow the basic Code Scaling Applicability and Uncertainty (CSAU) approach outlined in NUREG/CR-5249 (Reference 8). While the CSAU approach defines the process by which uncertainty analysis is performed, it leaves room for the applicant to determine the exact statistical methodology to be applied. In both the AOO application of TRACG and the ATWS application, GENE chose to apply a Normal Distribution One-Sided Upper Limit statistical methodology. The approach taken for application of TRACG to the ESBWR LOCA event is somewhat different.

Previous uses of the TRACG methodology have made use of Normal Distribution One-Sided Upper Tolerance Limit statistics to assess the uncertainty in the analyses. Application of the code to the ESBWR advanced passive system design relies on a very different approach to uncertainty since all calculations indicate the core remains covered and does not heat up. Uncertainty evaluation is done in this case using a much simpler [[ ]]. Staff concludes that this is acceptable since it is in accordance with the guidance given in Regulatory Guide 1.157, "Best-Estimate Calculations of ECCS Performance," May 1989 (Reference 35).

The CSAU methodology (Reference 8) states that an applicant should identify the important phenomena and rank them with respect to their effect on the safety criteria for the scenario. GEH provided a phenomena identification and ranking table (PIRT) that includes consideration of long-term core cooling for the ESBWR in Reference 4 and in Chapter 6, Section 6G, of the ESBWR DCD (Reference 7). For higher elevation breaks (i.e., MSLB and FWLB), the parameters that affect the long-term core cooling are the capacity of the GDCS pool relative to the reactor pressure vessel (RPV) volume, the heat removal capacity of the passive containment cooling system (PCCS) relative to the decay heat, and the condensation on drywell surfaces relative to the condensation in the PCCS. The phenomena that ranked high for the MSLB and FWLB are decay heat, GDCS pool volume versus elevation, and RPV volume versus elevation. GEH gave PCCS capacity a ranking of medium for these events.

For lower elevation breaks (i.e., GDLB and BDLB), the parameters that affect the long-term behavior are the capacity of the GDCS pool relative to the lower drywell volume, the pressure drop through the depressurization valves (DPVs), the heat removal capacity of the PCCS relative to decay heat, and to a smaller degree, the condensation on drywell surfaces relative to the condensation in the PCCS. The phenomena that ranked high for the BDLB and GDLB are 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.

For the higher elevation breaks, the RPV is filled to the break elevation during the re-flood phase. After the GDCS drains, the level is maintained by the PCCS drain flow which condenses the steam generated by the decay heat. These phenomena are not different from those reviewed by the staff for the short-term response and are documented in the staff SER and pre-application LTR (the approved staff safety evaluation and pre-application LTR for TRACG application) (Reference 1). The purpose of the PIRT ranking is to ensure that TRACG has the necessary models and qualification to simulate the important phenomena.

As discussed in the staff SER (Reference 1), the staff determined that TRACG04 has all of the necessary models and qualification to simulate this behavior. Therefore, the PIRT does not require revision at this time. If GEH proposes to use the rankings to select the uncertainties to be included in an uncertainty analysis in the future, a PIRT revision may be requested. The staff finds the GEH PIRT acceptable for demonstrating long-term core cooling of higher elevation LOCAs.

For the lower elevation breaks, the RPV is filled to the top of the chimney partition during the re-flood phase. After the GDCS drains, the level continues to drop to the level of the spillover holes in the drywell, and similar to the case of the higher elevation breaks, the level is maintained by the PCCS drain flow condensing the steam generated by the decay heat. These phenomena are not different from those reviewed by the staff for the short-term response and are documented in the staff SER and Pre-Application LTR (Reference 1). The size of the lower drywell becomes important during the long term.

As discussed in the staff SER (Reference 1), the staff determined that TRACG04 has all of the necessary models and qualifications to simulate this behavior; the PIRT does not require revision at this time. If GEH proposes to use the rankings to select the uncertainties to be included in an uncertainty analysis in the future, a PIRT revision may be requested. The staff finds the GEH PIRT acceptable for demonstrating long-term core cooling of lower elevation LOCAs.

Some condensation of the steam will occur in the drywell, which may affect the volume of water that will return to the vessel from the PCCS drainline. The TRACG ESBWR containment model was designed to maximize peak pressure and does not contain heat sinks, so it may not accurately calculate this condensation. Because of the large heat transfer area in the PCCS relative to that considered in the drywell, the staff concluded that the amount of condensation in the drywell should be small in comparison to the condensation in the PCCS.

In Request for Additional Information (RAI) 6.2-144 (Reference 9), the staff requested that GEH investigate the effects of assuming lower pressure in the drywell on RPV level calculations. In the RAI response the applicant provided the long-term post LOCA containment parameters which are within the design limits and therefore are acceptable to the staff. This study confirmed that this issue does not have a large impact on long-term core cooling. Based on the applicant's response, RAI 6.2-144 was resolved. Confirmatory Item 1 is closed.

2 Item 2: Break Spectrum and Core Uncovery

2.1 Confirmatory Item 2

> *"During the design certification review, the staff will verify that the TRACG application procedures conservatively calculate the collapsed water level in the chimney above the hot*

channel for the three break locations, MSLB [main steamline break], BDLB [bottom drainline break] and GDLB [gravity driven cooling system line break].

"Reference [NEDC-33083P, "TRACG Application for ESBWR," November 2002], Table 2.4-2 indicates that the GDLB results in the lowest static head in the chimney of the three break locations examined, the GDCS line, the main steam line, and the bottom drain line. At the design certification stage, GENE will need to provide supporting analyses for a spectrum of break locations to demonstrate that there is no core uncovery for the possible break locations. Should core uncovery occur, review of the TRACG code will be revisited to determine the adequacy of the applicable models and correlations.

"The procedures should be applicable to both short term and long term LOCA events (i.e., up to 72 hours)."

2.2 Staff Evaluation of Confirmatory Item 2

The LOCA analysis methodology in the Pre-Application LTR (applicant's submittal contained in the approved TRACG application for ESBWR) (Reference 1) was applied to minimum water level calculations for double-ended guillotine-sized breaks in the GDCS line, a bottom drainline, and a main steamline. This confirmatory item required that GEH perform a break spectrum analysis using break sizes and locations different from those used in the applicant's submittal contained in the Pre-Application LTR (Reference 1).

2.2.1 Other Break Locations

In response to staff RAI 6.3-46, GEH provided the results of analyses of additional break locations (Reference 10). In addition to the FWLB, MSLB, GDLB, and BDLB break results presented in DCD, Tier 2, (Reference 7), GEH listed the collapsed chimney levels for double-ended guillotine breaks in the GDCS equalizing line, the DPV stub tube/isolation condenser (IC), reactor water cleanup/shutdown cooling return line, and the IC return line. DCD Tier 2 showed that the most limiting cases are GDCS injection line and ICS drainline breaks. The applicant's results do not show heatup or core uncovery for any of the analyzed LOCAs. The applicant provided the full spectrum break analyses according to the guidance in SRP 6.3 and incorporated this information in DCD Revision 4, Section 6.3, which the staff found acceptable. Based on the applicant's response, RAI 6.3-46 was resolved.

In RAI 6.3-65 S01 (Reference 11), the staff also requested that GEH analyze a break in the standby liquid control system (SLCS) line and demonstrate that it is not the limiting break. The evaluation of RAI 6.3-65 S01 shows that an SLCS line break is not limiting. Based on the applicant's response, RAI 6.3-65 S01 was resolved.

2.2.2 Break Spectrum

In the SER for ESBWR, Chapter 6, Section 6.3.2.3.5, the staff evaluated the break spectrum. The staff concluded that GEH analyzed all vessel penetration break locations and break sizes, which resulted in analysis of the most limiting break. No new phenomena were introduced compared to the LOCA analysis listed in Licensing Topical Report (LTR) 33083-P-A.

2.2.3 Long-Term Core Cooling

The long-term core cooling PIRT evaluation, discussed in Confirmatory Item 1, is acceptable for the reasons noted above. On August 24, 2007, the staff received the response to RAI 6.3-79 (Reference 32), regarding the long-term core cooling analysis. GEH's response stated that for the ESBWR design, conformance to the requirement of adequate long-term cooling is assured and demonstrated for any LOCA where the water level can be restored and maintained at a level above the top of the reactor core. The response discussed TRACG calculation results for a short term (0 to 2,000 seconds) and a long term (0 to 72 hours). These calculations used assumptions with possible emergency core cooling system (ECCS) component single failures. (The FSER for Chapter 6, Section 6.3 provides a more detailed evaluation of RAI 6.3-79.)

The staff finds that this analysis demonstrates that the design provides for adequate long-term cooling. Based on the applicant's response, RAI 6.3-79 was resolved. The response to RAI 6.3-79 also closed one item included in RAI 21.6-98, which requests that GEH submit the long-term core cooling analysis.

In RAI 21.6-96 S01, the staff asked GEH to justify the TRACG model treatment of noncondensable (NC) gases, which effectively forces all of the air and noncondensable gases out of the drywell during a LOCA. The staff was concerned that this approach may not be conservative for long-term core cooling calculations, where the presence of non-condensibles in the PCCS would degrade the capability of the PCCS to condense steam and return inventory back to the vessel. The GEH response dated August 26, 2008 includes a discussion and a reference to RAI 21.6-69 S01 for the MSLB case as an example. GEH described that most of the NC gases in the drywell annulus are purged into the wetwell, and in the example case, the NC gas mass fraction entering into the PCCS is very low after a few hours. A comparison to the decay heat shows that the PCCS is over capacity in a few hours. During this over-capacity condition, the PCCS regulates the heat removal rate to match decay heat by accumulating NC gases in the lower part of the PCCS tube. A small increase in NC gas accumulation reduces the PCCS condensation capacity and will cause an increase in the drywell pressure. This drywell pressure increase will cause some NC gases in the lower part of the PCCS tube to be pushed through the PCCS vent into the wetwell. This increases the PCCS heat condensation capability, and equilibrium is reestablished with the drywell conditions.

The staff found the above GEH description of the phenomena reasonable. Acknowledging that the minimum RPV water level is reached in the earlier phase of a LOCA and that an adequate water inventory in the RPV is presented after GDCS injection, the staff agrees that the accuracy of NC gas modeling in the PCCS has little impact on the core cooling. The TRACG model ensures that most of the NC gases are purged into the wetwell, which is a conservative approach for peak containment pressure modeling. The staff considers the modeling of the NC gases to be acceptable in the long-term LOCA analysis. Based on the applicant's response, RAI 21.6-96 S01 Part A was resolved.

2.2.4 Conclusion

For all of the additional break locations and sizes that GEH simulates using TRACG for the LOCA analysis, including long-term core cooling, the ESBWR shows no heatup. Therefore, the staff does not need to revisit its review of the TRACG code to reexamine its application to core heatup in the ESBWR. Confirmatory Item 2 is therefore closed.

3 Item 3: Missing Definitions in TRACG Equations

3.1 <u>Confirmatory Item 3</u>

"GENE has committed to incorporate the missing definition for Ef, and new equations for the transition criterion between churned turbulent and annular flow, including the drift velocity term in updated code model description documentation."

3.2 <u>Staff Evaluation of Confirmatory Item 3</u>

The constitutive correlations for interfacial shear and heat transfer in TRACG are dependent on the flow regime in each hydraulic cell. Therefore, the flow regime for each cell must be identified before the flow equations are solved for that cell. 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.

When reviewing the pre-application LTR (Reference 1), the staff based its evaluation of the TRACG interfacial shear model on NEDE-32176P, Revision 2, "TRACG Model Description" (Reference 12). The staff requested additional information on the models describing the GEH calculation for transition to annular flow and the entrainment fraction. GEH had modified this model in TRACG04, and therefore, the description of the models in NEDE-32176P" (Reference 12) was not applicable to the version of TRACG being used to perform design calculations of the ESBWR. GEH submitted the updated models in NEDE-32176P, "TRACG Model Description", Revision 3 (Reference 13), in April 2006.

The models for flow regime transitions in TRACG02 had been qualified only at high pressure in NEDE-32177P, Revision 2, "TRACG Qualification" (Reference 16). GEH qualified TRACG against low-pressure data to extend the applicability of TRACG to LOCA applications. In TRACG04, GEH made changes to the model for transition from churned turbulent to annular flow to better match these data. The GEH criterion for transition to annular flow is when the liquid film can be lifted by the vapor flow relative to the liquid in the churn turbulent regime. This is satisfied at the void fraction where the same velocity is predicted for churn turbulent flow as it is for annular flow. GEH set the vapor velocity in the churn regime equal to that in the annular regime and solved for the transition void fraction. GEH modified the distribution parameter used to calculate the vapor velocity in the churn turbulent regime.

As described in Section 5.1.2 of NEDE-32176 (Reference 13), GEH also modified the entrainment model to better match the low-pressure data. An entrainment correlation developed by Mishima and Ishii is used in TRACG. GEH modified the model for entrainment in the case where only a fraction of the wall surface has gone into film boiling. GEH assumes that the liquid will flow only on the fraction of the wall that has not experienced boiling transition and can be wetted. The TRACG02 model uses a linear model that directly modifies the entrainment fraction in terms of the fraction of rod groups in boiling transition (Ef). The model in TRACG04 incorporates the wetted perimeter in the calculation of the hydraulic diameter in the entrainment correlation such that the entrainment fraction has a nonlinear relationship with the wetted perimeter. Both the TRACG02 and TRACG04 models have the correct limits in that, if there are no rod groups in boiling transition, there is no modification to the entrainment fraction.

If all rods are in boiling transition, in TRACG02, Ef goes to 1, which forces the entrainment fraction to be equal to 1; in TRACG04, as the wetted perimeter gets smaller, the hydraulic diameter goes to infinity, causing the entrainment fraction to be 1 since entrainment is calculated using a hyperbolic tangent (tgh(η)) dependency of the hydraulic diameter.

GEH further modified the Mishima and Ishii correlation based on the TRACG assessment against void fraction data. GEH found that the void fraction was overpredicted for conditions where there is a large entrainment fraction. GEH found that, since the entrainment was based on the hyperbolic tangent function (tgh(η)), it approached 1.0 too fast, causing the overprediction of the void fraction. GEH modified the tgh(η) functional dependence and kept the dimensionless property groups intact.

Figure 5-3 in Reference 13 shows the TRACG04 entrainment correlation compared to data. The correlation predicts well, with an average error in the entrainment fraction of +0.0008 and a standard deviation of 0.056.

The drift velocity used to calculate interfacial shear in the dispersed annular flow regime is based on the entrainment fraction. In RAI 21.6-75 (Reference 14), the staff requested that GEH submit the updated qualification report. In response, GEH submitted Revision 3 of the TRACG qualification report (Reference 15) in August 2007.

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, “TRACG Qualification, August 2007” (Reference 15) are very close to the results from NEDE-32177P, Revision 2, (Reference 16) which was assessed as satisfactory during the ESBWR preapplication phase of the design certification review. This ensures that the conclusion from the preapplication TRACG review is still valid. In addition, NEDE-32177P, Revision 3, increases the assessment cases to include Toshiba Low-Pressure Void Fraction Tests, Ontario Hydro Void Fraction Tests, and Centro Informzioni Studi Esperienze (CISE) Density Measurement Tests. The Toshiba tests were added to extend the qualification basis to lower pressures at 0.5 and 1.00 megapascal (MPa). The Ontario Hydro Void Fraction test results provide void fraction data for a large-scale pumped flow facility. The CISE Density Measurement test results provide data for void and quality relationships. 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. For these reasons, the staff concluded that the applicant’s response was adequate, and RAI 21.6-75 was resolved. Confirmatory Item 3 is closed.

4 Item 4: Update TRACG Model Description

4.1 Confirmatory Item 4

“The description of the TRACG model, Reference [NEDE-32176P, Rev. 2, “TRACG Model Description”, December 1999], will be updated to reflect all current models and correlations, thereby providing a level of detail consistent with a stand-alone document.”

4.2 Staff Evaluation of Confirmatory Item 4

In Revision 2 of NEDE-32176P, GEH removed the containment-related sections for the various models and correlations that had been included in Revision 1. GEH has returned this information to Revision 3 of NEDE-32176P. NEDE-32176P, Revision 4, "TRACG Model Description," issued January 2008 (Reference 19), supersedes NEDE-32176P, Revisions 2 and 3. (References 13 and 18).

GEH had also removed Section 7.11 ("Containment Components") from the "Component Model" section of NEDE-32176P in Revision 2. As a result, the information on drywell, wetwell air space, suppression pool, and main vents, such as that included in Table 6.5-3 in Revision 1, was not in Revision 2. GEH has returned Section 7.11 to Revision 3 of NEDE-32176P. However, the Revision 1 subsection "Model Assessment" was significantly shortened when it became Section 7.11.7.7, "Model Applicability," in Revision 3, by the removal of three figures (Figure 7.11-5, "Pressure Suppression Test Facility"; Figure 7.11-6, "Drywell Pressure Response"; and Figure 7.11-7, "Vent Flow Transient"), and the related details.

The staff considers these figures to be important, as they show the facility schematics and dimensions and compare the TRACG predictions with the measured drywell pressure and vent flow rate data. In RAI 21.6-107, the staff asked GEH to either justify the removal of these figures or include the figures as updated for the latest design. In the response to RAI 21.6-107, GEH stated that the removal of the figures is justified because they are related to TRACG assessment and not to TRACG model description. The staff disagreed with this response and requested in RAI 21.6-107 S01 that GEH replace the figures. In its response to RAI 21.6-107 S01, GEH committed to including the figures in a future submittal and did include Figures 7.11-6 and 7.11-7 in NEDE-33440P, Revision 1, "ESBWR Safety Analyses—Additional Information," issued June 2009 (Reference 17). The TRACG comparisons with experimental data previously reported in Figures 7.11-6 and 7.11-7 have been redone using more recent TRACG04 calculations. Figure 7.11-5 is available as Figure 5.5-1 in Reference 18. Based on the applicant's response, RAI 21.6-107 S01 was resolved.

When reviewing the pre-application LTR (Reference 1), the staff based its evaluation of the TRACG models and correlations on Revision 2 of the TRACG model description (Reference 12). This document is the basis for TRACG02. Since GEH uses TRACG04 for ESBWR licensing calculations, the staff requested that GEH submit an updated model description that reflects the models and correlations in TRACG04. GEH submitted Revision 3 to the TRACG model description in April 2006 (Reference 13) and submitted Revision 4 (Reference 19) in January 2008.

GEH submitted a list of the changes from TRACG02 to TRACG04 with its application to migrate the approved methodology for boiling-water reactor (BWR)/2–6 anticipated operational occurrence (AOO) and anticipated transient without scram (ATWS) overpressure analyses from TRACG02 to TRACG04 (Reference 19). A description of the differences between the versions and the staff's evaluation as applied to ESBWR LOCA analyses follows.

4.2.1 PANAC10 to PANAC11

TRACG02 is based on PANAC10 physics methods, whereas TRACG04 is based on those of PANAC11. Since ESBWR LOCA analyses do not use three-dimensional neutron kinetics, this change does not affect the staff's acceptance of TRACG for performance of ESBWR LOCA analyses as documented in the staff SER (Reference 1). However, Section 4.3 of the staff's SER on ESBWR design certification discusses the staff's review of PANAC11 applicability to ESBWR steady-state nuclear design. In addition, the safety evaluation (Reference 20) for NEDE-33083P, Supplement 3, "Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design," discusses a review of TRACG04 three-dimensional kinetics as applied to ESBWR transients.

4.2.2 Decay Heat Model

The American Nuclear Society decay heat model is implemented in TRACG04 as an optional model in addition to the existing May-Witt model. Since ESBWR LOCA analyses do not use the decay heat model in TRACG, this change does not affect the staff's acceptance of TRACG to perform ESBWR LOCA analyses as documented in the staff SER (Reference 1).

For ESBWR LOCA analyses, GEH takes decay heat values from a power table that is input into TRACG. The staff's review of the decay heat model as applied to ESBWR AOO analyses appears in Reference 20.

4.2.3 Quench Front Model

As part of TRACG04, GEH enhanced and activated the quench front model within the TRACG04 code. This model is used during the initialization of the re-flood phase of a LOCA. The staff has not reviewed this model; however, since the ESBWR does not experience heatup during a LOCA, the quench front model is not used. Therefore, this change does not affect the staff's acceptance of TRACG to perform ESBWR LOCA analyses as documented in the staff SER (Reference 1).

4.2.4 Hot Rod Model

GEH implemented a hot rod model in the TRACG one-dimensional thermal-hydraulic model of the channel component. This is to account for the thermal-hydraulic cross-sectional variations that lead to reduced heat transfer and higher fuel temperatures in certain rods. This model is used where peak cladding temperatures (PCTs) are calculated, such as during a LOCA. The staff did not review the hot rod model for LOCA application since the ESBWR does not experience heatup during a LOCA event. Therefore, this change does not affect the staff's acceptance of TRACG to perform ESBWR LOCA analyses as documented in the staff SER (Reference 1).

4.2.5 Minimum Film Boiling Temperature

The boundary between the transition boiling regime and the film boiling regime is defined by the minimum stable film boiling temperature. In addition to the Iloeje correlation and the homogeneous nucleation correlation, GEH implemented an additional option for calculating the minimum stable film boiling temperature, the Shumway correlation. The TRACG input decks used for the LOCA analyses use the Iloeje correlation. The staff has not reviewed the Shumway correlation and finds the use of the Iloeje correlation acceptable for ESBWR applications for LOCA, AOO, and ATWS. For LOCA and AOO events, the core does not enter film boiling, and therefore this correlation is not used. For ATWS events where the core does go into film boiling, the minimum stable film boiling temperature is used only to determine when the core will quench and has no effect on the value of the maximum PCT.

4.2.6 Entrainment Model

GEH modified the entrainment model in TRACG to better match low-pressure void fraction data for LOCA applications, as described in Section 3 of this report.

4.2.7 Flow Regime Map

GEH modified the transition from churn turbulent to annular flow models in TRACG to better match low-pressure void fraction data for LOCA applications, as discussed in Section 3 of this report.

4.2.8 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 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 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).

The GEH response to RAI 6.3-55 includes a description of the TRACG04 calculations, as discussed in the following paragraphs for RAI 6.3-54. The response to RAI 6.3-55 does not provide sufficient justification for combining models. However, the response to RAI 6.3-54 S01 addresses the impact of using gap conductance and fuel thermal conductivity from different models (GSTRM and PRIME03-based TRACG04, respectively) on TRACG04 calculations. Since this issue is being addressed in the supplements to RAI 6.3-54, the staff concludes that RAI 6.3-55 is 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 [[ ]]. 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 ESBWR, the NRC staff requested that GEH provide experimental data and benchmarks as well as TRACG02 (GSTRM) versus TRACG04 (PRIME03-based) thermal conductivity sensitivity study results in RAI 6.3-54 S01. In response to RAI 6.3-54 S01 (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 ESBWR.

GEH did not include LOCA sensitivity studies in response to RAI 6.3-54 S01 because the water level remains above the top of active fuel in ESBWR LOCA analyses. Consequently, there is no fuel heat up. Therefore, the impact of fuel thermal conductivity and gap conductance is much less significant than in cases where fuel heat-up 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-33083P).

NRC staff performed ESBWR LOCA fuel conductivity sensitivity confirmatory calculations 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. However, the fact remains that the PRIME03 code as well as the TRACG04 (PRIME03-based) thermal conductivity model have not been submitted for ESBWR application with the appropriate supporting empirical data. Therefore, future ESBWR TRACG LOCA analyses must be performed using the GSTRM model for both gap conductance and thermal conductivity, and the conclusions and limitations (including [[ ]] penalty) drawn by the NRC staff evaluation of GEH's Part 21 report (Appendix F to the SE for NEDC-33173P) (Ref. 33) are applicable to this SE. Should the NRC subsequently approve PRIME03 or another methodology for thermal conductivity and gap conductance for use with TRACG04 for ESBWR LOCA analyses, the fuel conductivity and gap conductance models must be consistent.

4.2.9 Cladding Perforation Models

GEH implemented models for the uncertainty in fuel rod internal pressure, the cladding yield stress, and the cladding rupture stress. GEH implemented these models for use in statistical analyses of a LOCA. The staff did not review these models since the ESBWR does not experience heatup during a LOCA and therefore does not invoke these models. In addition, GEH does not currently perform a statistical uncertainty analysis of ESBWR LOCA events. The staff finds that this change does not affect the staff's acceptance of TRACG to perform ESBWR LOCA analyses as documented in the staff SER (Reference 1).

4.2.10 Cladding Oxidation Model

GEH modified the cladding oxidation model to be consistent with the Cathcart and Pawel correlation. The staff did not review this model since cladding oxidation occurs at high temperatures. TRACG is not calculating heatup in any of the ESBWR LOCA events; therefore, this change does not affect the staff's acceptance of TRACG to perform ESBWR LOCA analyses as documented in the staff SER (Reference 1).

4.2.11 Enhanced Default Pump Homologous Curves

TRACG uses homologous curves to describe the pump head and torque response as a function of fluid volumetric flow rate and pump speed. GEH has supplemented the default pump homologous curves in TRACG04 with curves representative for large pumps. The ESBWR design does not credit pumps in performing LOCA analyses. However, GEH modeled PCCS vent fans in TRACG using a pump homologous head versus flow curve and provided points on this curve in ESBWR DCD Tier 2 Section 6.2. The pump homologous head versus flow curve provides minimum requirement for performance of PCCS vent fans. The staff used this curve as input in its confirmatory MELCOR containment LOCA analyses. Based on its review and confirmatory analysis, the staff determined that TRACG modeling of PCCS vent fans using a pump homologous head versus flow curve acceptable.

4.2.12 Improved Free Convection Heat Transfer

GEH implemented the McAdams correlation for free convection heat transfer used in drywell calculations. Section 15.2 of this report addresses the staff's evaluation of GEH's implementation of this correlation.

4.2.13 Optional Six-Cell Jet Pump

TRACG02 currently uses a five-cell jet pump model. TRACG04 has an option to subdivide the straight section between the suction inlet and the diffuser into two cells for a six-cell jet pump model. The ESBWR does not have jet pumps. Therefore, this change does not affect the staff's acceptance of TRACG to perform ESBWR LOCA analyses as documented in the staff SER (Reference 1).

4.2.14 Improved Boron Model

Boron is not modeled in ESBWR LOCA analyses; therefore, this change does not affect the staff's acceptance of TRACG to perform ESBWR LOCA analyses as documented in the staff SER (Reference 1). The safety evaluation of NEDE-33083P, Supplement 2, "Application of the TRACG Computer Code to Anticipated Transients Without Scram for the ESBWR Design" (Reference 27) discusses the application of the TRACG04 boron model to ESBWR ATWS analyses.

4.2.15 Revision 4 Evaluation

The changes made in Revision 4 of LTR NEDE-32176P (Reference 19) can be categorized into three types. The first type involves editorial changes, which have no impact on the code and its application. The second type includes changes to the model, and the third includes the changes related to the ESBWR modeling. This section presents evaluations of the second and third types of changes.

<u>Model Changes</u>

In Revision 4, GEH made the following changes to the model:

- change in the mass flux at which the Biasi correlation is used from 300 kilograms/square meter second (kg/(m^2s)) to 200 kg/(m^2s) (Section 6.6.6)

- change in values of constants xa and xb [[
  ]] (Section 9.5.1)

The staff discussed the two model changes with GEH. GEH stated that these changes are only documentary correction and do not involve any actual code changes. The staff verified the TRACG source code during an onsite review and found that the codes use the correct values.

The staff did not find any new changes to the source codes. Therefore, the staff does not consider these to be changes that affect the code performance and concludes that the documentary corrections are acceptable.

Modeling Change

In Revision 4, GEH made the following change in modeling:

- addition of a paragraph to discuss test data from PANDA pertaining to the wetwell gas space (Section 7.11.2)

GEH discussed how the test data from the PANDA facility show that the top of the wetwell gas space, which receives leakage flow from the drywell through the vacuum breakers, is at a higher temperature than the lower part of the gas space because of thermal stratification. [[

]] an irreversible frictional loss. The staff concurs that this approach produces conservatively high local gas space temperatures in the vicinity of the leakage and therefore is acceptable.

4.2.16 Conclusions

The staff has evaluated the changes in TRACG from TRACG02 and TRACG04. Many of the changes have no impact on the ESBWR LOCA calculations. For those changes that do affect the ESBWR LOCA calculations, the staff finds that the impacts are minor and acceptable.

5 Items 5 and 6: Isolation Condenser Testing

5.1 Confirmatory Items 5 and 6

> *"Further investigations are needed to conclusively determine the sound in the PANTHERS-IC testing that may have been due to water hammer, and to confirm its prevention in the ESBWR (e.g., by changing the hardware design of the IC [isolation condenser] inlet line or the startup procedure).*
>
> *"The PANTHERS-IC testing was terminated when leakages were detected in the IC upper header. As a result, the leakage issue was never resolved, and is an IC structural integrity issue that needs to be resolved for the ESBWR design certification."*

5.2 Staff Evaluation of Confirmatory Items 5 and 6

To prevent water hammer, GEH will control the slope of the condensate return line to avoid trapping of steam in the drain piping. In addition, GEH will control the rate of opening of the condensate return valves. The design of the isolation condenser system (ICS) is presented in DCD, Tier 2, Section 5.4.6 and was reviewed as part of the design certification review.

In Section 21.5.3 of the SER for the ESBWR design certification, the staff discussed its evaluation of the testing of the ICS at PANTHERS. GEH has agreed to perform power ascension tests to confirm the structural integrity of the ICS. GEH will also be able to confirm the possibility of water hammer during this testing. The staff therefore finds that successful completion of the IC startup testing, as described in Chapter 14 of the DCD (Reference 7), will adequately address Confirmatory Items 5 and 6.

6 Item 7: Scram Delay Time

6.1 Confirmatory Item 7

> *"During the design certification review stage, the ECCS baseline model should include the scram delay time and the 2 percent power measurement uncertainty."*

6.2 Staff Evaluation of Confirmatory Item 7

In the Summary of TRACG LOCA SER Confirmatory Items (Reference 3), GEH stated that it has included the scram delay and 2-percent power uncertainty in the DCD, Chapter 6 analyses. However, the time at which the scram occurs is different from that discussed in the pre-application LTR (Reference 1) in response to pre-application RAI-324. For the analyses presented in the ESBWR DCD (Reference 7), GEH assumed a scram upon initiation of the break. The scram occurs with the loss of power assumption coincident with the break. GEH incorporated 2 seconds of delay time because of the signal delay. GEH accounted for the travel time in the rods in the decay heat curve. GEH submitted these details in response to RAI 6.3-52 (Reference 21). The staff received the RAI response on December 21, 2007. GEH noted that the scram time delay used in the TRACG input decks for the LOCA events described in DCD, Tier 2, Chapter 6, is 2.25 seconds. This delay time was incorporated into the DCD, Tier 2, Chapter 6, LOCA TRACG input decks through a TRIP card. This total time delay of 2.25 seconds is based on and justified by the following partial time delays:

- 2.00 seconds for sensor delay
- 0.05 seconds for sensor trip scram solenoid to deenergize (reactor protection system logic)
- 0.20 seconds for scram solenoid deenergized rods to start to move (scram valve open)

Based on the applicant's response, RAI 6.3-52 was resolved.

DCD, Tier 2, Table 6.3-11, "Plant Variables with Nominal and Bounding Calculation Values" documents the 2 percent reactor power uncertainty which is included in the DCD TRACG calculations. Since this uncertainty has been included in the calculations, the staff considers Confirmatory Item 7 to be closed.

7 Item 8: Additional Detail in TRACG Modeling

7.1 Confirmatory Item 8

> *"During the design certification stage, separate modeling of the vessel shield, the reflective thermal insulation layer, and the air gap from the lumped heat structure will be necessary."*

7.2 Staff Evaluation of Confirmatory Item 8

In response to this confirmatory item, GEH stated that separate modeling of the vessel shield, the reflective thermal insulation layer, and the air gap from the lumped heat structure was added to the TRACG model (Reference 3). The staff evaluated the modeling documents at the GEH site during an onsite review trip and confirmed that the TRACG model includes the required input. Confirmatory Item 8 is satisfied.

8 Item 9: Chimney Nodalization Studies

8.1 Confirmatory Item 9

> *"Nodalization studies will be necessary at design certification to calculate the minimum water level in the chimney partition."*

8.2 Staff Evaluation of Confirmatory Item 9

During the preapplication phase of the TRACG for ESBWR LOCA review, GEH and the NRC staff investigated the effect of nodalization and bundle power distributions on the calculated minimum water level in the chimney during a LOCA in the ESBWR. The core and chimney region are represented by three concentric rings in the TRACG input deck. GEH performed studies varying the radial peaking factors in the bundles feeding the three rings in the core. GEH found that, when all of the bundles in Ring 1 (the innermost ring in the TRACG input deck) are set to the highest radial peaking factor (with the two outer rings reduced accordingly), the difference in minimum level calculated by the three separate rings could vary by about [[ ]]. GEH stated that, because of the "drafting" effect (i.e., enhanced two-phase flow and heat transfer in the hot ring because of additional two-phase driving head in the chimney), this modeling strategy would be nonconservative (see RAI-329 and RAI-406 in pre-application LTR (Reference 1).

The staff agreed with GEH that the drafting effect would make this modeling strategy nonconservative. The staff performed independent calculations in an attempt to reduce the drafting effect by creating a smaller chimney partition above a smaller number of hotter bundles in Ring 1. The staff found that this modeling strategy reduced the minimum static head in the chimney. The staff concluded that the nodalization presented in Figure 2.7-1 in pre-application LTR (Reference 1) is adequate for calculating the core-average minimum chimney water level during an ESBWR LOCA.

GEH submitted in a letter the Summary of TRACG LOCA SER Confirmatory Items (Reference 3) to address the staff's confirmatory items related to the SER on TRACG as applied to an ESBWR LOCA. In this letter, GEH stated that it had addressed this item and that the nodalization includes individual chimney partitions. In addition, the staff requested in RAI 21.6-98 that GEH provide all TRACG nodalization changes and that the five chimneys used to calculate minimum water level be identified. In the response to RAI 21.6-98 (Reference 25), GEH explained that two individual chimneys are added to the three super chimneys that represent each of the three rings. The staff reviewed the TRACG input decks submitted by GEH and determined that GEH had added two individual chimney partitions to the ESBWR vessel. These are represented by [[ ]] components, with one located above each of the two hot channels. GEH uses these components to calculate the collapsed liquid level in the

ESBWR chimney. The staff found that the revised nodaliztion described in the applicant's response to RAI 21.6-98 adequately represents the ESBWR reactor vessel. Subsequent DCD TRACG analyses have been based on this refined vessel model, so the staff considers RAI 21.6-98 resolved. This closes Confirmatory Item 9.

9 Item 10: Treatment of Loss of Feedwater

9.1 Confirmatory Item 10

> *"The assumption of the loss of feedwater flow used by GENE is not conservative. Therefore, the existing GENE MSLB model and the current analysis approach underestimate the maximum containment pressure and temperature. At the design certification phase, this should be resolved."*

9.2 Staff Evaluation of Confirmatory Item 10

In RAI 21.6-103, the staff requested that GEH address this confirmatory item. GEH responded to RAI 21.6-103 in a letter dated April 24, 2009 (Reference 26). In this response, GEH added a feature to isolate ESBWR feedwater following a LOCA on high-high drywell pressure.

Other features available to isolate feedwater following a LOCA include (1) high feedwater differential pressure coincident with high drywell pressure, (2) high drywell pressure coincident with lower drywell high water level, (3) reactor low-low water level with a 1-hour time delay, and (4) reactor high water level (ESBWR DCD, Tier 2, Section 5.4.5.3.3). GEH also added features to mitigate the effect of another outside water source that automatically initiates in LOCA events, high-pressure injection mode of the control rod drive system (HP CRD): (1) HP CRD makeup isolation signal on two out of three GDCS pool low level and (2) HP CRD makeup isolation signal on drywell water level high coincident with drywell pressure high. In addition, GEH made design changes to increase the containment margin for LOCA events: (1) raising the drywell to suppression chamber spillover hole 0.5 meters (m), which reduces the amount of high-temperature inventory added to the suppression pool during breaks below reactor normal water level and (2) changing technical specification maximum allowable operating drywell pressure to 15.5 pounds per square inch absolute (psia), which reduces the mass of NC gas in the containment. GEH performed containment pressurization analysis after making the above changes.

DCD, Tier 2, Chapter 6, (Reference 7) documents the bounding MSLB cases with loss of feedwater (LOFW) and without LOFW. The LOFW case is illustrated in Reference 24, Table 6.2-7g and Figures 6.2-14f and 6.2-14g. Reference 24, Table 6.2-7h and Figures 6.2-14j and 6.2-14k illustrate the no-LOFW case. A comparison of the figures shows almost identical containment temperature and pressure results for the two runs. In both cases, containment pressure following a LOCA would stay below the containment design value for 72 hours.

By making the above changes, GEH addressed the staff's concern about the assumption of the LOFW flow during containment analysis, since the results with or without feedwater available are shown to be almost identical and the containment design pressure is not exceeded in either case. Based on the applicant's response, RAI 21.6-103 was resolved. This closes Confirmatory Item 10.

10 Item 11: Feedwater Heater Modeling

10.1 Confirmatory Item 11

"Without detailed feedwater heater system design information, both the staff and GENE had to make assumptions about the mass and energy discharge from the feedwater heater system. The staff believes that the bounding containment peak pressure and temperature need to be evaluated during the design certification stage after the feedwater heater system design is finalized. If the evaluation indicates that the code application range is exceeded or a new scenario, such as wetwell flooding, has not been examined during the pre-application stage, the staff may choose to review the TRACG code for such new use."

10.2 Staff Evaluation of Confirmatory Item 11

As stated in the ESBWR preapplication SER "TRACG Application for ESBWR," the staff was concerned about an assumption that the feedwater pump is tripped and the feedwater flow is lost after an MSLB accident (see staff SER (Reference 1)). Although this led to a conservative PCT evaluation as it reduces the available coolant inventory, the assumption is nonconservative for containment analysis. The feedwater carrying the feedwater heater train stored energy increases the mass and energy discharge through the break into the containment leading to higher containment pressures and temperatures.

GEH included modeling of the feedwater line system in the TRACG analysis. ESBWR DCD, Tier 2, Figure 6.2-8b, shows the TRACG nodalization of the ESBWR feedwater line system. In addition, as discussed in Section 9 of this report, GEH added features to isolate feedwater after a LOCA. For the containment analysis, GEH assumed a continued flow of feedwater into the containment until its isolation. This addresses the staff's concern because feedwater is isolated following an MSLB accident. This closes Confirmatory Item 11.

11 Item 12: Address Power Transient Resulting from Main Steam Isolation Valve Closure

11.1 Confirmatory Item 12

"The quick closure of the MSIVs while control rods are being inserted may increase the total core power due to void collapse. At the design certification stage, GENE should evaluate the effects of void collapse for the GDCS and BDLB LOCA cases."

11.2 Staff Evaluation of Confirmatory Item 12

GEH stated that there is no significant power transient because of void collapse from the main steam isolation valve (MSIV) closure effect since the control rods are always inserted before the MSIVs close for all of the breaks. The staff was able to confirm this upon review of the ESBWR LOCA analyses. The staff agrees with the GEH assessment and concurs that this is not an issue for the ESBWR LOCA event. This closes Confirmatory Item 12.

12 Item 13: Assess TRACG against Some Standard Containment Problems

12.1 Confirmatory Item 13

> *"During the staff's earlier review of the SBWR [simplified boiling-water reactor], work that GENE relies on for the ESBWR, the staff noted that GENE had not evaluated more traditional integral containment tests such as the Marviken tests, the Carolinas Virginia Tube Reactor test 3 without sprays, and the Battelle-Frankfurt Model Containment tests C-13 and C-15, for MSLBs. In response to staff RAI 317.1, GENE agreed to perform assessments of TRACG to model containment performance against integral test data that is publicly available for International Standard Problems where the test facilities and tests are well defined. The tests to be analyzed will be specified later, and the analysis will be completed during the design certification review.*

"The staff also requested that GENE provide a plan and schedule to assess the ability of TRACG to model containment performance against additional separate effects tests. Separate effects tests that should be considered include the Wisconsin Flat Plate condensation tests, (References … [I.K. Huhtiniemi and M.L. Corradini, "Condensation in the Presence of Noncondensible Gases", Nuclear Engineering Design, 141, pp. 429-446, 1993; M. Siddique, "The Effects of Noncondensible Gasses on Steam Condensation Under Forced Convection Conditions," MIT, January 1992; and K. Lian, "Experimental and Analytical Study of Direct Contact Condensation of Steam and Water", MIT, May 1991]).

In response to staff RAI 317.2, GENE agreed to perform assessments of TRACG to model containment performance against separate effects test data that is publicly available for International Standard Problems where the test facilities and tests are well defined. The tests to be analyzed will be specified later, and the analysis will be completed during the design certification review."

12.2 Staff Evaluation of Confirmatory Item 13

In RAIs 21.6-98 and 21.6-103, the staff requested that GEH address this confirmatory item. The response to RAI 21.6-103 (Reference 26) states that the response to RAI 21.6-98 addresses Confirmatory Item 13. The response to RAI 21.6-98 dated August 29, 2008 (Reference 25) includes two standard problems. Attachments A and B of the response include TRACG simulation results for the integral Marviken blowdown test 18 and the Wisconsin Flat Plate separate effect condensation tests. The staff's review of Reference 25 finds the TRACG simulation results to be acceptable because of the good agreement with the test results. This information is also included in LTR NEDE-33440P, Revision 1 (Reference 17), as referenced in the ESBWR DCD, Tier 2, Revision 6, Reference 6.2-11.

GEH performed a comparison of the TRACG simulation results with the Marviken test data for the short term (0 to 4.4 seconds) and the long term (0 to 220 seconds). The purpose was to assess TRACG's capability to predict a vent clearing transient (short term), steam/air transport through the vent system (long term), and containment pressure and temperature responses (short and long term). The staff reviewed this comparison and concluded that considering the measurement uncertainties, TRACG calculations agree well with the Marviken test data. General trends were predicted successfully.

GEH also evaluated the TRACG capability to predict the Wisconsin Flat Plate steam condensation data obtained in the vertical position of the test section in the presence of NC gases. Measured average condensation heat transfer coefficients were not sensitive to the plate inclination angle. Two different condensation correlations were assessed in a one-dimensional TRACG nodalization model of the vertical pipe simulating a PCCS section. Even though both correlations overpredicted the test data by a widely varying degree, the ESBWR post-LOCA peak drywell pressure is not sensitive to the choice of correlation. This is because, during a LOCA in the ESBWR, most of the NC gas is displaced to the wetwell gas space, and the NC gas mass fraction near the drywell wall is very small. The staff agrees with the GEH assessment, since it is consistent with observations from past tests, and therefore, Confirmatory Item 13 is considered closed.

13 Item 14: Gravity-Driven Cooling System Gas Space and Wetwell Vent Modeling

13.1 Confirmatory Item 14

"GDCS gas space and the wetwell vent should be modeled correctly during the design certification stage."

13.2 Staff Evaluation of Confirmatory Item 14

In DCD, Tier 2, Revision 5, GEH changed the ESBWR design so that the GDCS gas space is now connected to the drywell. GEH stated in Reference 3 that it would submit all TRACG nodalization changes related to NRC SER confirmatory Items.

In addition, the response to RAI 21.6-98 notes that the changes to TRACG nodalization are discussed in Sections 6A and 6B of the ESBWR DCD (Reference 24). The staff reviewed the detailed comparison provided in DCD Table 6A-1 between the original TRACG model described in the approved version of NEDC-33083P-A (Reference 1) and the revised TRACG model, which reflects the changes in design. DCD, Tier 2, Appendix 6B provides a description of the GEH evaluation of the differences in the LOCA results using the revised TRACG model and the original design results. The staff determined that all significant model parameters were addressed, and that the design changes were appropriately modeled. The detailed TRACG containment and RPV nodalization diagrams provided in DCD Figures 6A-1 and 6A-2, respectively, were also evaluated by the staff and determined to be sufficiently refined to represent the ESBWR design. This closes Confirmatory Item 14.

14 Item 15: Add Detail to the Containment Portion of the Emergency Core Cooling System Evaluation Model

14.1 Confirmatory Item 15

"During the design certification review, if the ECCS evaluation model is used beyond 2000 seconds, additional VESSEL levels need to be added on top of the existing [[ ]], and the pool needs to be modeled in the same fashion as is done for containment/LOCA modeling."

14.2 Staff Evaluation of Confirmatory Item 15

GEH combined the containment and ECCS evaluation (RPV level) model into one model for the design certification, and therefore, the additional levels have been added to the containment for the ECCS evaluation (RPV-level calculations). Although GEH has combined the two input decks, they are slightly different, as certain assumptions are needed to make each analysis conservative. GEH submitted the differences between the two input decks in response to RAI 6.3-45 (Reference 27). For the bounding calculations, GEH maintained the conservative assumptions that the staff previously reviewed in Section 2.7.2.1 for vessel water level and Table 3.7-1 of for peak pressure in the pre-application LTR (Reference 1). Because of design changes and error corrections, GEH has implemented nodalization changes, which were evaluated by the staff during an audit of TRACG as applied to an ESBWR LOCA. In RAI 21.6-98, the staff requested that GEH formally submit these changes to the staff. In the response to RAI 21.6-98 (Reference 25), GEH noted that Sections 6A and 6B of the ESBWR DCD (Reference 24) discuss the changes to TRACG nodalization.

Some nodalization changes made to the TRACG model for calculating peak containment pressure were not implemented for the TRACG model that calculates long-term core cooling. The changes implemented in the TRACG model for calculating peak pressure are spillover holes, higher intake elevation for the GDCS drainpipes, two vent paths between the GDCS air space and the drywell versus one vent path, and a fine nodalization of the PCCS vent line. GEH stated, and the staff agrees, that the effect of these items occurs at a later stage of the transient, and therefore, these changes will have no impact on minimum water level for the LOCA. However, the staff issued a supplement to RAI 6.3-45 requesting that GEH justify its contention that, even though the input deck for calculating minimum water level lacks the modifications applied to the containment input deck, it still provides accurate or conservative results for the long-term core cooling analysis.

The staff received the response to RAI 6.3-45 on June 20, 2007, and the response to RAI 6.3-45 S01 on March 25, 2008. GEH responded that the model differences described in the response to RAI 6.3-45 have been reconciled in the analyses of DCD, Tier 2, Revision 4. A consistent set of assumptions, the same TRACG model, and a consistent input deck have been used to calculate minimum water levels and perform containment peak pressure of nominal cases. However, different assumptions were made for the bounding cases between containment analysis and RPV water-level analysis. GEH updated the table in the response to RAI 6.3-45 and identified the differences for the bounding cases. These differences include normal water level in the downcomer and suppression pool.

Because of the conservative minimum initial water level, the staff agrees with GEH that using the lower water level is bounding for the LOCA analysis minimum water-level calculation and using the higher water level in the suppression pool (SP) is bounding for the peak containment pressure calculation. GEH clarified the difference between the minimum water level calculation and the peak containment pressure analyses. Based on the applicant's response, RAI 6.3-45 was resolved.

GEH combines the TRACG model for the containment peak pressure evaluation with that of the RPV minimum water level calculation. This is inconsistent with the approved methodology in the pre-application LTR (Reference 1), which states that "the drywell model [is] set to minimize containment pressurization rate." In RAI 6.2-144, the staff asked GEH to justify the use of the containment model in calculating minimum water level. In response, GEH evaluated the impact of containment back pressure on the ECCS performance and presented this evaluation in DCD, Tier 2, Revision 4, Appendix 6C (Reference 28). 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. Confirmatory Item 15 is closed.

15 Item 16: [[ ]] Factors

15.1 Confirmatory Item 16

"Prior to submission of the final design analyses in support of design certification, GENE should perform a review of the appropriateness of the [[ ]] factors and the liquid/vapor interface heat transfer used in the containment modeling."

15.2 Staff Evaluation of Confirmatory Item 16

[[ ]] factors are used to account for the way in which the presence of NC gases reduces the interfacial heat transfer.

In NEDE-32176P, Revision 3, GEH made the following modifications to address this item:

- GEH previously used the Holman correlation (Equation 6.5-28) to model the interfacial heat transfer at the suppression pool free surface. A sensitivity study by GEH found that the TRACG model results were not very sensitive to the Holman correlation. However, the staff was concerned that the conclusion was not valid for all possible situations. GEH explored more correlations and found the McAdams correlation to be more general. NEDE-32176P, Revision 3, Section 6.5.8, includes a detailed description of the McAdams, Grashof, and Prandtl numbers-based free-convection correlation for flat plates (Equation 6.5-51). The McAdams correlation is the default model for the interfacial heat transfer at a free surface, though Holman's simplistic expression can still be selected via the user input.

- GEH has also included additional details in Section 6.5.8 of NEDE-32176P, Revision 3, to describe the Sparrow-Uchida degradation factor that accounts for the reduction of the interfacial heat transfer due to the presence of the NC gases. GEH has replaced Figure 6.5-1 from NEDE-32176P, Revision 1, with Figure 6-13 in Revision 3.

  The new figure not only shows the composite Sparrow-Uchida curve shown on Figure 6.5-1 (Revision 1) that TRACG uses, but also the individual Uchida and Sparrow curves that were independently developed for the high and low NC gases-to-steam ratios, respectively. In Section 6.5.8.2, GEH has added a description of how the composite Uchida-Sparrow data are implemented within the TRACG code.

- GEH has expanded Section 6.5.8.3 in NEDE-32176P, Revision 3, to explain the applicability of the McAdams and Holman correlations for a variety of conditions. While the Holman correlation is applicable to turbulent flow ($GrPr > 10^9$) (Gr and Pr are Grashof and Prandtl numbers) only, the McAdams correlation is applicable to a much wider range ($10^5 < GrPr < 3x10^{10}$). The discussion also addresses the effect of heat transfer enhancement due to interfacial ripples and the uncertainties in the Sparrow-

Uchida degradation factor and the Kuhn-Schrock-Peterson (K-S-P) correlation and their interrelation.

The staff reviewed the applicability of the [[ ]] correlation to the ESBWR interfacial heat transfer at the pool interface. The staff acknowledges that interfacial heat transfer, in general, is a complex phenomenon and the available physical models are subject to substantial uncertainties. Since the sensitivity study described in NEDE-32176P, Revision 3 indicates that this phenomenon (i.e., degradation of heat transfer at the pool surface due to noncondensable gases) has a relatively small effect on the peak containment pressure the staff finds the TRACG interfacial heat transfer at the pool interface to be acceptable for ESBWR design certification analyses.

The staff concludes that GEH has provided sufficient explanation of the range of the applicability of the correlations and hence, Confirmatory Item 16 is closed.

16 Item 17: GEH Assurance that TRACG Models and Correlations Are Consistent with the Final ESBWR Design

16.1 Confirmatory Item 17

"Prior to performing the final design analyses at the design certification stage, GENE should perform a thorough evaluation of the ESBWR design records and TRACG ESBWR model development records to substantiate that the TRACG models and correlations are consistent with the final design requirements and intended application."

16.2 Staff Evaluation of Confirmatory Item 17

GEH stated in Reference 3 that the design records for ESBWR and TRACG model development are consistent with the GEH quality assurance (QA) system and that the application range of the correlations in the final design is within the reviewed application range. The NRC staff performed an audit of TRACG as applied to ESBWR LOCA analyses and was able to evaluate the GEH QA processes and their application to TRACG development and use for ESBWR design certification. The staff confirmed that GEH has rigorous QA processes for TRACG04A and that TRACG04A is being applied within its application range for ESBWR LOCA applications. The staff found that TRACG04P is being used for some licensing calculations. The staff issued RAI 21.6-95 and RAI 21.6-96 to address the open items associated with the audit of TRACG04P.

The staff received the response to RAI 21.6-95 from GEH on November 19, 2007. RAI 21.6-95 requested that GEH address the changes to TRACG04 from Versions 42 to 45. In GEH's response, it summarized the changes in TRACG04 from Versions 42 to 45. GEH claimed that these changes have been demonstrated to have no or minimal impact on the calculated ESBWR and operating BWR results. The staff confirmed this statement by review of the GEH calculations during the QA audit (Reference 36). The staff therefore considers RAI 21.6-95 to be closed.

The staff received the response to RAI 21.6-96 on June 21, 2007, and the response to RAI 21.6-96 S01 on August 26, 2008. RAI 21.6-96 requested that GEH clarify the TRACG code version used in DCD Chapters 4, 6, and 15 and compare the results from TRACG04A (ALPHA VMS version) to the results from TRACG04P (PC version). GEH provided the versions used in

DCD Chapters 4, 6, 15 and the comparisons of the key parameters between TRACG04A and TRACG04P. In its response to RAI 21.6-96, GEH stated that the differences between TRACG04 ALPHA and PC were caused partly by the inability of TRACG to accurately predict NC gas distributions in general. GEH used a conservative approach to minimize the long-term containment pressure sensitivity to NC gas concentrations. This conservative approach entailed modification of the input nodalization to force all NC gases out of the drywell. Using this approach, GEH was able to reduce the predicted long-term containment pressure differences between ALPHA and PC to < 1.0 percent.

After reviewing the response to RAI 21.6-96, the staff requested more information in RAI 21.6-96 S01, which has two parts. Part 1 requested that GEH address the conservatism in the NC gas assumption on the long-term core cooling water level in the RPV. The second part requested qualification of the code version used for the LOCA analysis.

GEH provided the qualification for TRACG04P in the responses to RAIs 21.6-96 S01 and 21.6-96 S02 to address the second part of this supplemental RAI.

GEH acknowledged that TRACG cannot accurately predict NC gas distributions in general and that a conservative approach was used, which minimizes the long-term pressure response sensitivity to NC gas concentrations by modifying the input model nodalization to force all the air out of the drywell. The staff remained concerned that this approach may not be conservative for long-term core cooling, since the presence of NC gases in the PCCS would degrade the capability of the PCCS to condense steam and return inventory to the vessel. GEH states that the PCCS is over capacity at about 3 hours. Under this condition, the PCCS regulates the heat removal rate to match the decay heat through the feedback between heat removal, condenser pressure, and NC gas holdup in the condenser. If additional heat removal is needed to condense the steam generated by decay heat, the pressure in the condenser would be high, which would increase the flow of NC gases out of the condenser to the vent. The decreased amount of NC gases in the condenser tubes would result in an increase in heat removal. The staff agrees that the NC gases assumption used does not result in nonconservative PCCS modeling for long-term core cooling because under this condition, the PCCS heat removal rate always matches the decay heat rate through one of the two operating modes. If the steam condensation is established, the PCCS heat exchangers can remove more steam than that generated by the actual level of decay heat. If, due to the NC gas collection, the PCCS condensation rate decreases, the DW pressure increases to the point when drywell to wetwell pressure difference (delta-P) exceeds the submergence of the PCCS vent pipe (without clearing the main horizontal vents), establishing the flow to the suppression pool and removing the NC gas to the wetwell gas space. This “delta-P” mode of operation re-establishes steam flow from the DW and its condensation in the PCCS. These two operating modes, i.e., condensing and delta-P, are the essential design features of the PCCS self-regulating operation. The staff determined that the TRACG04P (PC version) is capable of analyzing both PCCS modes of operation and therefore is acceptable for calculating the long-term containment pressure. Based on the applicant’s response, RAI 21.6-96 S02 was closed.

Therefore, Confirmatory Item 17 is closed.

17 Item 18: Uncertainty Analysis

17.1 Confirmatory Item 18

"At the design certification stage, GENE should examine further whether or not an uncertainty analysis can be performed on the combined reactor coolant system/containment system calculation rather than treating the containment aspect of the ECCS LOCA calculation in a bounding way. The uncertainty analysis methodology should be applicable to both short term and long term LOCA events (i.e., up to 72 hours)."

17.2 Staff Evaluation of Confirmatory Item 18

In the Summary of TRACG LOCA SER Confirmatory Items (Reference 3), GEH states that since there is no core heatup, an uncertainty analysis of PCT would not provide useful results. GEH states that a bounding evaluation for the minimum water level in the chimney during a LOCA event would demonstrate that there is margin to core uncovery and heatup.

10 CFR 50.46(a)(1)(i)) (Reference 29) states, in part, that "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…." The regulation in 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." GEH has not selected either of these options. The staff issued RAI 6.3-81 requesting that GEH demonstrate how the LOCA analyses comply with this requirement.

The staff received the RAI response on January 25, 2008. GEH responded 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 889 degrees Celsius (C) (1,600 degrees Fahrenheit (F)) to the limit of 1204 degrees C (2,200 degrees F) (acceptance criteria set forth in paragraph (b) of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors"). GEH further noted that the static head inside the chimney (in meters of water) is selected as the figure of merit for comparison and for use in 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. The TRACG model parameter uncertainties and plant parameter uncertainties have been identified (GEH LTR NEDC-33083P-A, Sections 2.4 and 2.5.3). 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 m to +0.2 m (GEH LTR 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 that there is 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. The staff concludes that GEH's LOCA results comply with the requirement of 10 CFR 50.46. Therefore, based on the applicant's response, RAI 6.3-81 was closed.

18 Item 19: Passive Containment Cooling System Vent System

18.1 <u>Confirmatory Item 19</u>

> *"The actual design configuration of the PCCS vent system, especially the vent submergence, may influence the amount of steam condensed in the SP. Therefore, during the design certification review, the staff will confirm that steam entering the SP through the PCCS vent, as designed, will perform as expected to condense steam entering the SP."*

18.2 <u>Staff Evaluation of Confirmatory Item 19</u>

The preapplication SER "TRACG Application for ESBWR" (NEDC-33083P) reports the following under PIRT item WW3, on page 39: "Based on available test data, GENE concluded that any steam entering the SP through the PCCS vent, based on the design presented for this review, will be condensed within the SP during the blowdown period of the accident."

The staff's acceptance of the above statement during the preapplication phase was based on its review of the supplemental information provided by GEH in its response to preapplication RAI 314.1 (Reference 30) regarding the PCCS performance during the blowdown. However, the staff was aware that even though the plant design has changed since the preapplication (e.g., change in power level from 4,000 megawatts (MW) to 4,500 MW, possible use of spargers), the same 0.9-m PCCS vent submergence depth as specified in NEDE-32176P, Revision 1, also appears in Revision 3.

In RAI 21.6-106, the staff asked GEH to confirm that the 0.9-m submergence depth is still valid and that the final PCCS vent design would adequately condense steam and lead to saturated steam, and not superheated steam, above the suppression pool. In response to RAI 21.6-106 and Supplements 1 through 3, GEH noted that the power increase from 4,000 MW to 4,500 MW is a 12.5-percent increase in power, which leads to a corresponding increase in PCCS vent steam mass flow rate. The number of PCCS vents was increased from four to six, resulting in an increase in vent area of 50 percent. Therefore, these changes result in a decrease of steam mass flow rate through each PCCS vent by 25 percent, which is conservative. The addition of spargers would enhance steam condensation. Therefore, the same submergence length of 0.9 m stated in NEDE-32176P, Revision 1, continues to be bounding, and the PCCS vent system would adequately condense steam and would lead to saturated steam and not superheated steam above the suppression pool. GEH's validation of the vent line design performance during the blowdown is based on experimental data. GEH performed a dimensional analysis of the condensation data for steam discharged through the PCCS vent in the LINX test facility (Reference 18). The test data showed that the steam was fully condensed in all tests that include a range of steam flow rates. Therefore, the staff determined that steam entering the SP through the PCCS vent, as designed, will perform as expected to condense steam entering the SP. GEH has documented these design changes in Chapter 6 of the ESBWR DCD, Tier 2, Revision 6. In addition, Table 6B-2 in the ESBWR DCD, Tier 2, Revision 6, contains a comparison of the design modeled in the ESBWR DCD analyses and in the original model development LTR, NEDE-32176P, which was not updated to reflect all the modeling design changes.

The staff finds the GEH justification and documentation to be sufficient. Therefore, based on the applicant's response, RAI 21.6-106 S03 was closed. Confirmatory Item 19 is closed.

19 Item 20: ESBWR Design Changes

19.1 Confirmatory Item 20

> *"This safety evaluation is based on the 4000 MWth ESBWR reference design as described in Reference … [NEDC-33084P, Revision 1, "ESBWR Design Description," August 2003]. At the design certification stage, GENE should demonstrate that the reference design as described in Reference … [NEDC-33084P, Revision 1] has not been altered in such a way as to affect the staff's conclusions of this report. Significant changes in the design that challenge the conclusions of this report will result in the staff reevaluating the applicability of the TRACG code."*

19.2 Staff Evaluation of Confirmatory Item 20

In ESBWR DCD, Tier 2, Revision 6, Chapter 6, Table 6G-2, and the response to RAI 21.6-98, GEH listed the changes to the ESBWR design between the design referenced during the preapplication review and the design submitted in the design certification application. The following subsections describe and evaluate the applicability of the staff SER and pre-application LTR (Reference 1) to each of the major changes.

19.2.1 Core Power

The preapplication power level was 4,000 megawatts thermal (MWt). The ESBWR, as described in Revision 3 of the DCD (Reference 24), has a core power level of 4,500 MWt. The higher power level will result in higher core exit and chimney void fractions. In RAI 21.6-75 (Reference 14), the staff requested that GEH submit the updated qualification report. In response, GEH submitted Revision 3 of the TRACG qualification report (Reference 15) in August 2007. 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 qualification of void fraction prediction is evaluated in Confirmatory Item 3 and is judged to be satisfactory. Based on the applicant's response, RAI 21.6-75 is closed.

19.2.2 Number of Bundles

The number of bundles was increased from 1,020 to 1,132 to accommodate the power uprate described in Section 19.2.1 of this report. The flexible input of TRACG allows GEH to change the number of bundles. This change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.3 Change in Core Shroud Size

The size of the core shroud was increased to include the bundles added (see Section 19.2.2 of this report). This causes the downcomer volume to decrease and therefore provides less inventory during the blowdown phase of the LOCA. GEH included additional ECCS sources to provide more inventory. Although this change affects the results of the analysis, it does not affect the ability of TRACG to simulate the analysis, since the TRACG input is flexible enough that GEH can change the size of the shroud within the TRACG input deck. This change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.4 Core Lattice

GEH changed the ESBWR control blade lattice from an F-lattice with wide blades to an N-lattice with standard blades. The purpose of this change was to simplify the design, as the N-lattice is similar to the current BWR/2–6. The TRACG LOCA model in the Pre-Application LTR (Reference 1) does not model three-dimensional kinetics and therefore does not consider the geometry of the control blades. GEH uses a decay heat table upon reactor scram.

During an audit from December 11 through December 15, 2006 and resuming for the period between December 19 and December 20, 2006, the staff reviewed the decay heat curve used in current ESBWR LOCA analyses in detail and confirmed that this change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.5 Number of Control Rod Drives

GEH increased the number of control rod drives from 121 to 269 to accommodate the N-lattice (see Section 19.2.4 above). Control rod drives are not modeled in TRACG ESBWR LOCA analyses; therefore, this change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.6 Gravity-Driven Cooling System Pool and Airspace Location

To simplify the ESBWR design, GEH changed the location of the GDCS pool airspace from the wetwell to the drywell. This is the same configuration as in the simplified boiling-water reactor (SBWR) and the M-series PANDA tests (Reference 31). The staff reviewed the PANDA M-series tests during its evaluation of the Pre-Application LTR Reference 1. This change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.7 Passive Containment Cooling System

GEH increased the number of PCCS units from four to six, reduced the heat removal capability (from 13.5 MW to 11 MW) of each PCCS unit, and credited PCCS vent fans to force the flow of steam through the PCCS after 72 hours following a LOCA. GEH performed full-scale tests of the PCCS, and the staff finds these tests applicable to the ESBWR design since the condenser tube diameter, length, and pitch are the same as those tested. The only difference is in the number of tubes. Section 21.5.3 of the SER for the ESBWR design certification discusses the staff's evaluation of the PCCS testing program. The staff included the PCCS vent fans in its confirmatory MELCOR analysis.

19.2.8 Isolation Condenser System

GEH increased the power level of each IC from 30 to 33.75 MWt. The staff did not evaluate the ability of TRACG to model the ICS during its evaluation of the pre-application (Reference 1); therefore, changes to the design do not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1). The staff reviewed the GEH testing of the ICS, which is discussed in Section 21.5.3 of the SER for the ESBWR design certification. The SER for NEDE-33083P, Supplement 3, Revision 1 (Reference 20) discusses modeling of the ICS for transient analysis.

During the preapplication review of the 4,000-MWt design, the staff did not evaluate the capability of TRACG to model the ICS because the system was not part of the ECCS at that time, and the GEH analyses took no credit for ICS operation during a LOCA. The ICS has been added to the ECCS for the updated 4,500-MWt ESBWR design by providing additional liquid inventory upon opening of the condensate return valves to initiate the system.

In DCD Chapter 6, Table 6A1, GEH stated that the initial water inventory in the ICs is modeled in the analysis, and no credit is assumed for the heat transfer in the ICs. TRACG is able to model additional IC inventory, and therefore, TRACG is adequate to model ICS in the LOCA analysis.

19.2.9 Pressure Relief System

GEH changed 12 automatic depressurization system (ADS) valves to 10 ADS valves and 8 safety/relief valves (SRVs) in the latest design. The TRACG critical flow model is independent of the number of valves. Therefore, this change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.10 Containment Vents

The suppression pool (wetwell) is connected to the drywell by a vent system. The number of vents was increased from 10 to 12. During the first part of a LOCA caused by a break in the drywell, a differential pressure is created from the drywell to the wetwell, and much of the gas and steam will be transferred to the wetwell through the vent system. The staff based its acceptance of the TRACG model of the containment vents in the staff SER (Reference 1) on comparisons of TRACG to the pressure suppression test facility (PSTF) facility for the 5703 series tests. These tests were performed for full-scale vents. TRACG was able to model the vent flow rates and time of vent clearing adequately. The change in number of vents reduced the mass flow rate in the vents, which is still within TRACG application range, and is acceptable.

19.2.11 Feedwater System

GEH changed the control logic on the feedwater system. The feedwater system is isolated during a feedwater line break due to high drywell pressure. The LOCA analyses for containment in the ESBWR DCD (Reference 24) assume alternating current (AC) power is available and the feedwater system is running. If the feedwater line break is assumed, more mass and energy is released to the containment. It is conservative to assume the availability of AC power. The staff evaluated this change in logic and determined that it is conservative, and thus this change is acceptable.

19.2.12 Turbine Bypass Capacity

GEH increased the turbine bypass capacity from 33 percent to 110 percent. This change is not modeled in TRACG ESBWR LOCA analyses and thus does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER Reference 1.

19.2.13 Passive Containment Cooling Drain Tanks

GEH removed the passive containment cooling (PCC) drain tanks that were once located in the drywell. Instead, the PCCS drains directly to the GDCS. This change simplifies the ESBWR design and represents the same configuration as in the SBWR and the M-series PANDA tests (Reference 31). The staff reviewed the PANDA M-series tests during its evaluation of the pre-application LTR (Reference 1). This change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.14 Suppression Pool Volume

GEH changed the suppression pool (SP) volume from 3,610 m^3 to 4,424 m^3. The larger suppression pool reduces the temperature increase in the pool. The TRACG input is flexible and capable of changing this design parameter. This change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.15 Drywell/Wetwell Volume Ratio

The drywell to wetwell volume ratio increased from 1.31 to 1.33. The TRACG geometry input is flexible and capable of changing this design parameter. This change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.16 Lower Drywell Free Volume to Top of Active Fuel Elevation

GEH reduced the volume of the lower drywell to the top of the active fuel elevation. This improves the performance for lower elevation breaks, such as the BDLB. For the long-term cooling performance for the lower elevation breaks, GEH relies on the drywell filling to an elevation above the top of active fuel so that the PCCS has to supply only enough water to compensate for the inventory losses in the core that result from steaming from decay heat. The TRACG geometry input is flexible and capable of changing this design parameter. This change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.17 Standby Liquid Control System Activated on Automatic Depressurization System

GEH added the SLCS to the ECCS to provide additional inventory during a LOCA. The SLCS is modeled as a [[ ]] table. The SLCS injection velocity was calculated for ESBWR ATWS (high-pressure) conditions and therefore will be bounding for use in ESBWR LOCA analyses. In letter MFN 07-312, dated June 2007 (Reference 27) GEH discusses the adequacy of using the SLCS FILL table for ATWS events. This change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.18 Isolation Condenser System Inline Vessel

GEH added one 9-m^3 vessel in each ICS train to improve the RPV water level in the LOCA. Since no new phenomena were introduced, this change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.19 Safety/Relief Valve Capacity

GEH increased SRV capacity by about 11 percent. Since no new phenomena were introduced, this change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.20 Feedwater Isolation Valve Configuration

GEH changed five valves per line to four process-operated valves per line. Since no new phenomena were introduced, this change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.21 Main Steamline Changes

GEH increased the main steamline diameter from 700 millimeters (mm) to 750 mm upstream of MSIV and pipelines of DPVs on ICs. This change does not introduce new phenomena and does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.22 Turbine Main Steam Piping Diameter

GEH changed the turbine main steam piping diameter from 800 mm to 750 mm. This change does not introduce new phenomena and does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.23 Main Steam Isolation Valve Size

GEH changed the MSIV size from 771 mm to 762 mm. This change does not introduce new phenomena and does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.24 Passive Containment Cooling System Vent Fan

GEH added one PCCS ventilation fan to each PCCS vent line, which ends submerged in the GDCS pool. This change enhances the PCCS condensation, but it does not introduce new phenomena. This change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.25 Drywell Spray Flow

GEH changed the spray flow rate from 3785 liters per minute (1,000 gallons per minute (gpm)) to 2120 liters per minute (560 gpm). The change introduces no new phenomena and does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.26 Cross-Tie between FAPCS and RWCU

GEH added a cross-tie from the fuel and auxiliary pool cooling system (FAPCS) suction line to reactor water clean up (RWCU) train A, upstream of the nonregenerative heat exchangers. This change does not affect the staff's evaluation of TRACG as applied to an ESBWR LOCA in the staff SER (Reference 1).

19.2.27 Conclusions for Confirmatory Item 20

GEH provided all of the design changes that impact the LOCA analysis since the approval of TRACG for the ESBWR LOCA analysis (NEDO-33083-A) in the ESBWR DCD, Tier 2, Revision 5. The impacts of these changes on the LOCA analyses have been reanalyzed and documented in Sections 6.2 and 6.3 in the ESBWR DCD, Revision 6. As evaluated in Sections 19.2.1 through 19.2.28 of this report, the justifications for the TRACG model updates provided by GEH are acceptable. Therefore, Confirmatory Item 20 is closed.

20 Conclusions

The staff reviewed the additional data provided by the applicant in final approved LTR (Reference 1) to address the remaining open items. The staff finds that the open items have been adequately addressed, and they are now closed.

The staff concludes that the TRACG code and methodology described in the Pre-Application LTR (Reference 1) and associated RAI responses are applicable to the calculation of an ESBWR LOCA as described in Sections 6.2 and 6.3 of the ESBWR DCD.

21 References

1. NEDC-33083P-A, "TRACG Application for ESBWR," March 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML051390265))[2]; Non-Proprietary version: NEDO-33083-A, "TRACG Application for ESBWR," October 2005, Accession No. ML053320203); Original submittal: NEDC-33083P, "TRACG Application for ESBWR," November 2002 (ADAMS Accession No. ML023260440).

2. "Summary of Exit Meeting Held on December 15, 2006, to Discuss Staff's Audit of TRACG Loss-of-Coolant Accident Analyses," January 4, 2007 (ADAMS Accession No. ML0635403880).

3. Letter from D. H. Hinds (GEH) to NRC, MFN 05-096, "Summary of September 9, 2005 NRC/GE Conference Call on TRACG LOCA SER Confirmatory Items," September 20, 2005 (ADAMS Accession No. ML052910378).

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

5. Letter from D. H. Hinds (GEH) to NRC, MFN 05-109, "GE[H] Responds to Results of NRC Acceptance Review for ESBWR Design Certification Application—Item 2 (TAC No. MC8168)," October 20, 2005 (ADAMS Accession No. ML053000054).

6. Letter from D. H. Hinds (GEH) to NRC, MFN 06-094, "Revised Response—GE[H] Response to Results of NRC Acceptance Review for ESBWR Design Certification Application—Item 2," March 28, 2006 (ADAMS Accession No. ML060900097).

[2] The Proprietary Staff Safety Evaluation Report is contained in this document on pages 11 through 185. The Non-proprietary version of the Staff Safety Evaluation is contained in NEDO-33083-A (ML053320203) on pages 6 through 178. The original proprietary submittal of the topical report is NEDC-33083P (ML023260440).

7. Letter from D. H. Hinds (GEH) to NRC, "General Electric Company—ESBWR Standard Plant Design Revision 7 to Design Control Document Tier 2," March 29, 2010 (ADAMS Accession No. ML101340143).

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

9. Letter from J. C. Kinsey (GEH) to NRC, MFN 07-310, "Response to Portion of NRC Request for Additional Information Letter No. 85—Containment Systems and Emergency Core Cooling Systems—RAI Numbers 6.2-144, 6.2-145, 6.2-146, 6.2-147, and 6.3-66," June 7, 2007 (ADAMS Accession No. ML071770542).

10. Letter from J. C. Kinsey (GEH) to NRC, MFN 07-049, "Response to Portion of NRC Request for Additional Information Letter No. 68—Emergency Core Cooling Systems—RAI Numbers 6.3-46 through 6.3-49," March 20, 2007 (ADAMS Accession No. ML070860262).

11. Letter from J. C. Kinsey (GEH) to NRC, "Response to Portion of NRC Request for Additional Information Letter No. 85 Related to ESBWR Design Certification Application—Emergency Core Cooling Systems—RAI Number 6.3-65 S01," March 25, 2008 (ADAMS Accession No. ML080870229).

12. NEDE-32176P, Rev. 2, "TRACG Model Description," December 1999 (ADAMS Accession No. ML993630283).

13. NEDE-32176P, Revision 3, "TRACG Model Description," April 2006 (ADAMS Accession No. ML061160238). NEDO-32176, Revision 4, January 2008 (ADAMS Accession No. ML080370271).

14. Letter from M. C. Barillas (NRC) to D.H. Hinds (GEH), "Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application," October 10, 2006 (ADAMS Accession No. ML062790238).

15. NEDE-32177P, Rev. 3, MFN 07-452, "Transmittal of GEH Topical Report, NEDE-32177P, Revision 3, *TRACG Qualification,* August 2007," August 29, 2007 (ADAMS Accession No. ML072480007).

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17. NEDE-33440P, Revision 1, "ESBWR Safety Analyses—Additional Information," June 2009 (ADAMS Accession No. ML091830295).

18. NEDC-32725P, Revision 1, "TRACG Qualification for SBWR – Document Transmittal for Pre-Application Review of ESBWR," MFN 02-053, August 30, 2002 (ADAMS Accession No. ML022560081)

19. NEDO-32176, Rev. 4, "TRACG Model Description," January 2008 (ADAMS Accession No. ML080370271).

20. Safety Evaluation for "Application of the TRACG Computer Code to the Transient Analysis for the ESBWR Design," NEDE-33083P, Supplement 3, Revision 1 (ADAMS Accession Nos. ML102430487, ML102560359, ML100270024).

21. Letter from S. A. Williams (NRC) to D. H. Hinds (GEH), "Request for Additional Information Letter No. 68 related to ESBWR Design Certification Application," October 10, 2006 (ADAMS Accession No. ML062770002).

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24. "General Electric Company—ESBWR Standard Plant Design—Revision 6 to Design Control Document Tier 2, Chapter 6 (Engineered Safety Features)," November 2009 (ADAMS Accession No. ML092680561).

25. Letter from R. E. Kingston (GEH), MFN 08-545, "Response to Portion of NRC Request for Additional Information Letter No. 85—Related to ESBWR Design Certification Application—RAI Number 21.6-98," August 29, 2008 (ADAMS Accession No. ML082480507)

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27. Letter from J. C. Kinsey (GEH) to NRC, MFN 07-312, "Response to Portion of NRC Request for Additional Information Letter No. 68—Containment Systems and Emergency Core Cooling Systems—RAI Numbers 6.2-98, 6.3-45, and 6.3-51," June 20, 2007 (ADAMS Accession No. ML071920093).

28. ESBWR Design Control Document Tier 2, Revision 4 (ADAMS Accession No. ML072900480).

29. Title 10, Energy, Part 50, "Domestic Licensing of Production and Utilization Facilities."

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31. MFN 04-059, "Update of ESBWR TRACG Qualification for NEDC-32725P and NEDC-33080P Using the 9-Apr-2004 Program Library Version of TRACG04," June 2004 (ADAMS Accession No. ML041610035).

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