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Reference: Letter S. Hucik, GE to S. Collins, NRC, Pre-application Review of ESBWR, dated April 18, 2002

Subject: **TRACG Application for ESBWR, NEDO-33083 - Document Transmittal for Pre-Application Review of ESBWR – Non-Proprietary Report**

The CD accompanying this letter contains the GE non-proprietary report NEDO-33083, TRACG Application for ESBWR.

If you have any questions about the information provided here, please contact Atam Rao at (408) 925-1885, or myself.

Sincerely,

C.J.W. Deacon

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TRACG Application for ESBWR

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT PLEASE READ CAREFULLY

This work was performed partially as part of a contract between various utilities and GE for "ESBWR Development".

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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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Staff Position	GE Process	Evaluation
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
[Redacted]		
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.
[Redacted]		

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.

[

Redacted

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]

4.4 Statistical Treatment of Overall Calculational Uncertainty

Staff Position	GE Process	Evaluation
[	Redacted	]
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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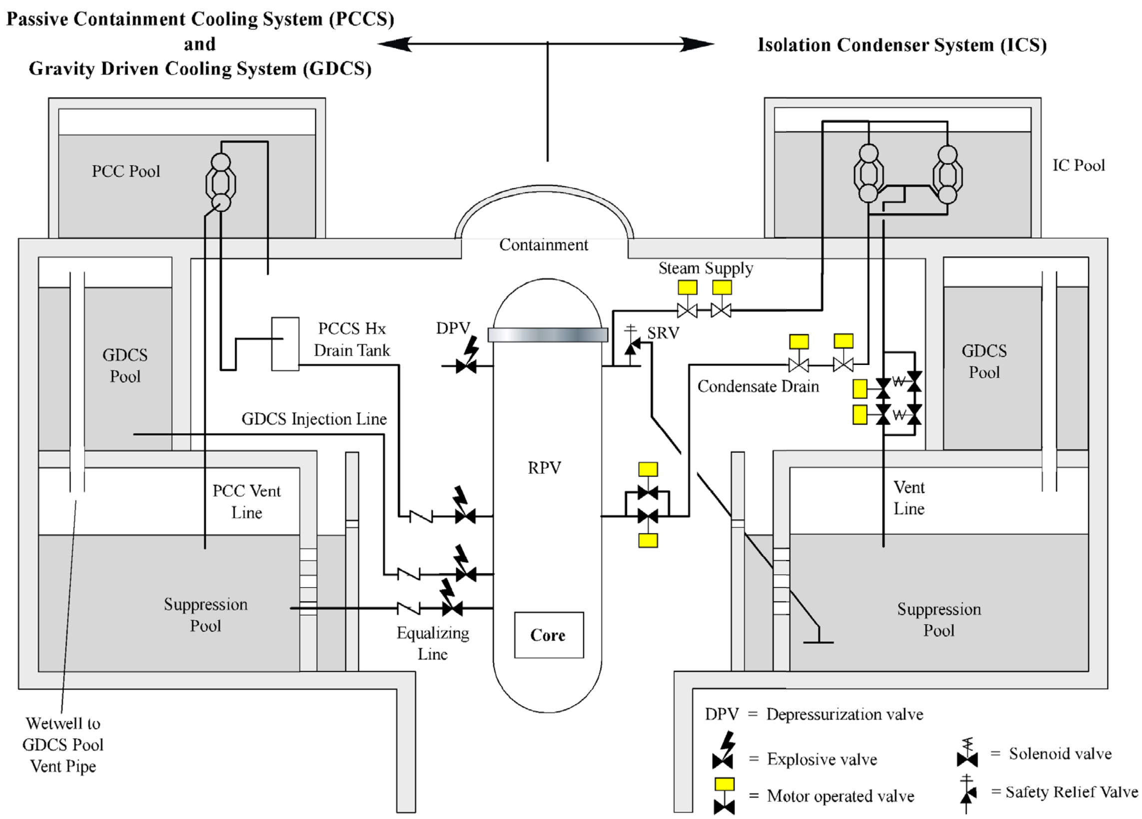

Figure 2.2-1 ESBWR Passive Safety Features

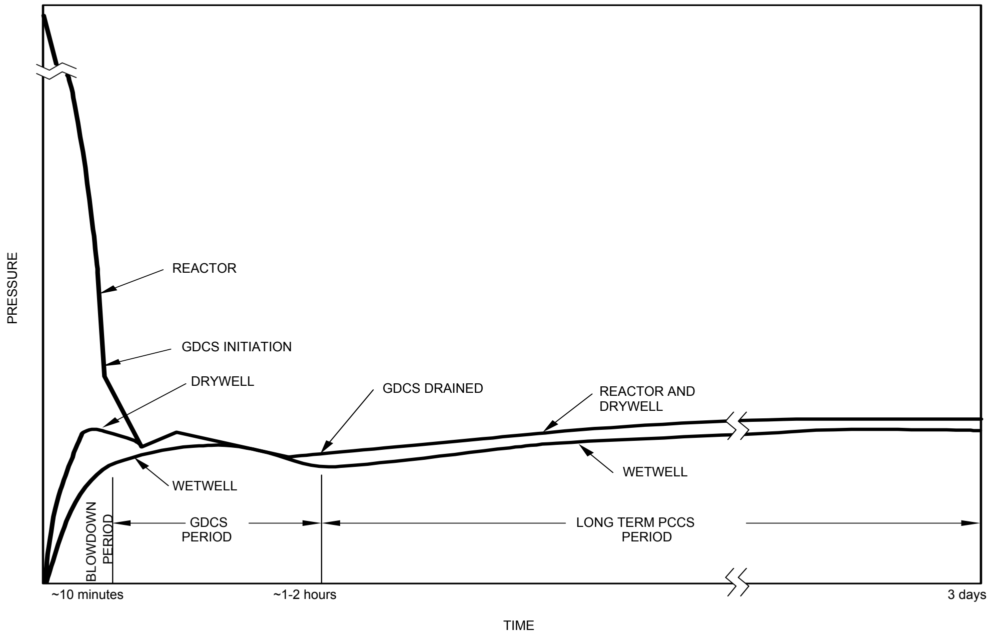

Figure 2.2-2 Phases of the LOCA Transient

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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.

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2.4.2 Effects of Nodalization

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2.4.3 Effects of Scale

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2.4.3.1 RPV Level

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2.4.3.2 Scale-up of Level Changes

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2.4.4 Sensitivity to PIRT Parameters

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2.4.4.1 Scoping ECCS/LOCA Break Spectrum Analysis

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2.4.4.2 Sensitivity Study on PIRT Parameters

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2.4.4.3 Sensitivity Study on Interactions between ECCS and Containment

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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

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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.

2.5.3 Uncertainties in Plant Parameters/Initial Conditions

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2.5.4 Sensitivity to Plant Initial Conditions

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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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2.6.1 Specific Application Process

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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.

2.7.1.1 TRACG Nodalization for ESBWR ECCS/LOCA Analysis

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2.7.1.2 Results for Baseline ECCS/LOCA Analysis

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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)

2.7.2 Bounding ECCS/LOCA Analysis

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2.7.2.1 Analysis Assumptions

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2.7.2.2 Bounding Analysis Results

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

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

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

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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 not 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 <u>not</u> 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

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Table 3.2-1 (Contd.)

Highly Ranked PIRT Phenomena for ESBWR Containment/ LOCA

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Table 3.2-1 (Contd.)

Highly Ranked PIRT Phenomena for ESBWR Containment/ LOCA

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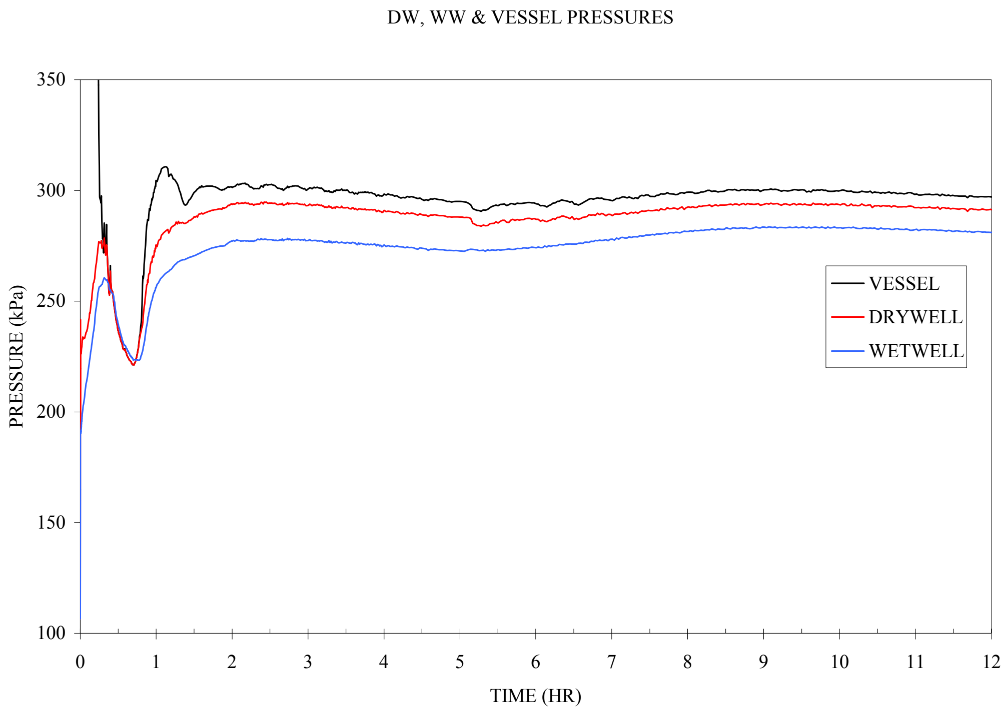

Figure 3.2-1 : Main Steam Line Break Vessel and Containment Pressures (Typical)

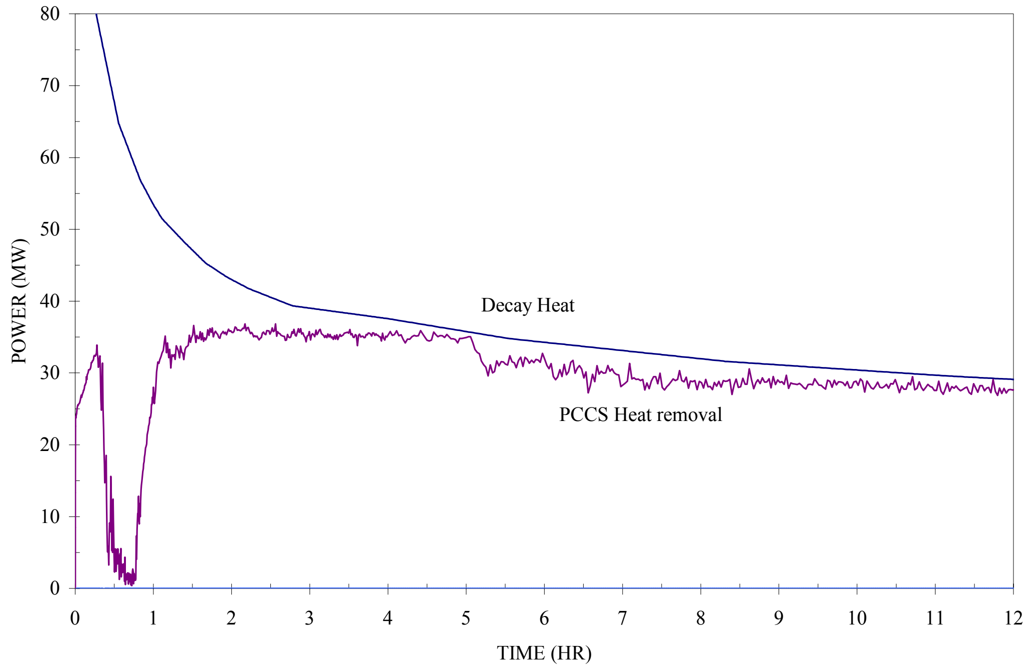

Figure 3.2-2 Main Steam Line Break Decay Heat and PCCS Heat Removal (Typical)

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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3.3.1.1.1 Suppression Pool Stratification

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3.3.1.1.2 Wetwell Gas Space Stratification

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3.3.1.1.3 Drywell Stratification

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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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3.4.1 Model Parameters and Uncertainties

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3.4.2 Effect of Scale

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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

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

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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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4.3 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.

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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4.4.2 Effects of Nodalization

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4.4.3 Effects of Scale

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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

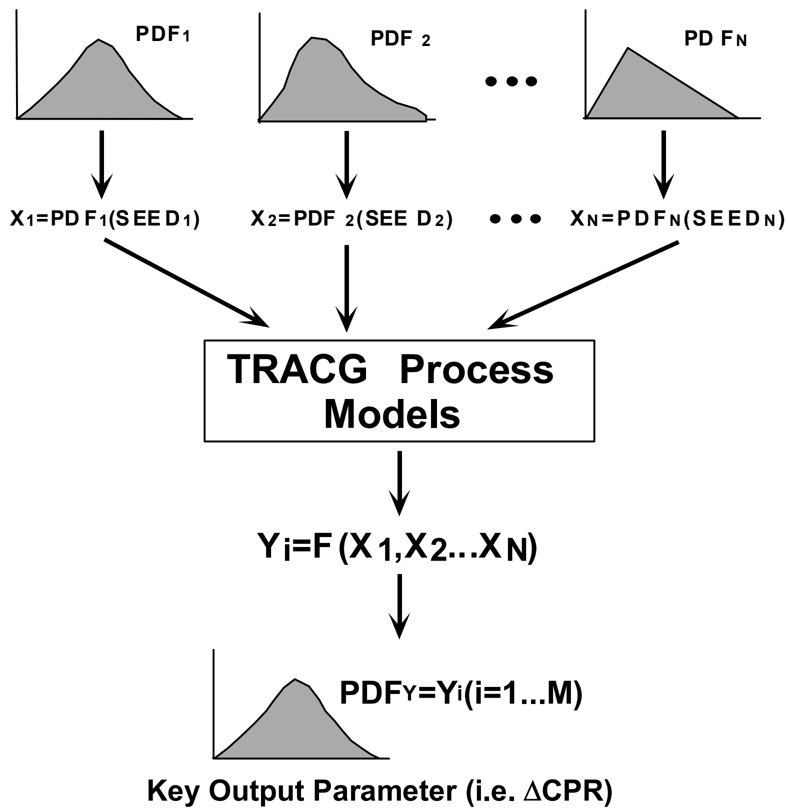

Figure 4.6-1. Schematic Process for Combining Uncertainties

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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4.6.2.1 Conformance with Design Limits

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4.6.3 Determination of OLMCPR

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4.6.3.1 Details of Process of OLMCPR Calculation

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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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4.7.1.1 Load Rejection No Bypass (LRNB) Baseline Analysis

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4.7.1.2 Feedwater Controller Failure (Maximum Demand at 150% of Rated)

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4.7.1.3 Main Steamline Isolation Valve (MSIV) Closure

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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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