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BEST-ESTIMATE CALCULATIONS OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE

A. INTRODUCTION

Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," of 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that light-water nuclear reactors fueled with uranium oxide pellets within cylindrical zircaloy cladding be provided with emergency core cooling systems (ECCS) that are designed in such a way that their calculated core cooling performance after a postulated loss-of-coolant accident (LOCA) conforms to certain criteria specified in paragraph 50.46(b). Paragraph 50.46(b)(1) requires that the calculated maximum temperature of fuel element cladding not be greater than 2200°F. In addition, paragraphs 50.46(b)(2) through (b)(5), which contain required limits for calculated maximum cladding oxidation and maximum hydrogen generation, require that calculated changes in core geometry remain amenable to cooling and that long-term decay heat removal be provided.

On September 16, 1988, the NRC staff amended the requirements of § 50.46 and Appendix K, "ECCS Evaluation Models" (53 FR 35996), so that these regulations reflect the improved understanding of ECCS performance during reactor transients that was obtained through the extensive research performed since the promulgation of the original requirements in January 1974. Paragraph 50.46(a)(1) now permits licensees or applicants to use either Appendix K features or a realistic[1] evaluation model. These realistic evaluation models[2] must include sufficient supporting justification to demonstrate that the analytic techniques employed realistically describe the behavior of the reactor system during a postulated loss-of-coolant accident. Paragraph 50.46(a)(1) also requires that the uncertainty in the realistic evaluation model be quantified and considered when comparing the results of the calculations with the applicable limits in paragraph 50.46(b) so that there is a high probability that the criteria will not be exceeded.

This regulatory guide describes models,[3] correlations,[4] 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 loss-of-coolant accident and for estimating the uncertainty in that

[1]For the purpose of this guide, the terms "best-estimate" and "realistic" have the same meaning. Both terms are used to indicate that the techniques attempt to predict realistic reactor system thermal-hydraulic response. Best-estimate is not used in a statistical sense in this guide.

[2]The term "evaluation model" refers to a nuclear plant system computer code or any other analysis tool designed to predict the aggregate behavior of a reactor during a loss-of-coolant accident. It can be either best-estimate or conservative and may contain many correlations or models.

[3]The term "model" refers to a set of equations derived from fundamental physical laws that is designed to predict the details of a specific phenomenon.

[4]The term "correlation" refers to an equation having empirically determined constants such that it can predict some details of a specific phenomenon for a limited range of conditions.

calculation. Methods for including the uncertainty in the comparisons of the calculational results to the criteria of paragraph 50.46(b), in order to meet the requirement that there be a high probability that the criteria would not be exceeded, are also described in this regulatory guide. Paragraph 50.46(a) also permits licensees to use evaluation models developed in conformance with Appendix K.

Other models, data, model evaluation procedures, and methods will be considered if they are supported by appropriate experimental data and technical justification. Any models, data, model evaluation procedures, and methods listed as acceptable in this regulatory guide are acceptable in a generic sense only and would still have to be justified to the NRC staff as being appropriately applied and applicable for particular plant applications.

The regulatory position in this regulatory guide lists models, correlations, data, and model evaluation procedures that the NRC staff considers acceptable for realistic calculations of ECCS performance. 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 Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

Any information collection activities discussed in this regulatory guide are contained as requirements in 10 CFR Part 50, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 50 have been cleared under OMB Clearance No. 3150-0011.

B. DISCUSSION

The criteria set forth in § 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and the calculational methods specified in Appendix K were promulgated in January 1974 after extensive rulemaking hearings and were based on the understanding of ECCS performance available at that time. In the years following the promulgation of those rules, the NRC, the nuclear industry, and several foreign institutions have conducted an extensive program of research that has greatly improved the understanding of ECCS performance during a postulated loss-of-coolant accident. The methods specified in Appendix K were found to be highly conservative; that is, the fuel cladding temperatures expected during a loss-of-coolant accident would be much lower than the temperatures calculated using Appendix K methods. In addition to showing that Appendix K is conservative, the ECCS research provided information that allows for quantification of that conservatism. The results of experiments, computer code development, and code assessment allow more accurate calculations, along with reasonable estimates of uncertainty, of ECCS performance during a postulated loss-of-coolant accident than is possible using the Appendix K procedures.

It was also found that some plants were being restricted in operating flexibility by limits resulting from conservative Appendix K requirements. Based on the research performed, it was determined that these restrictions could be relaxed through the use of more realistic calculations without adversely affecting safety. The Appendix K requirements tended to divert both NRC and industry resources from matters that are relevant to reactor safety to analyses with assumptions known to be nonphysical.

In recognition of the known conservatisms in Appendix K, the NRC adopted an interim approach in 1983, described in SECY-83-472,[5] to accommodate industry requests for improved evaluation models for the purpose of reducing reactor operating restrictions. This interim approach was a step in the direction of basing licensing decisions on realistic calculations of plant behavior. Although the approach permits many "best-estimate" methods and models to be used for licensee submittals, it retains those features of Appendix K that are legal requirements.

The current revision of § 50.46 permits ECCS evaluation models to be fully "best-estimate" and removes the arbitrary conservatisms contained in the required features of Appendix K for those licensees wishing to use these improved methods. Safety is best served when decisions concerning the limits within which nuclear reactors are permitted to operate are based upon realistic calculations. This approach is currently being used in the resolution of almost all reactor safety issues (e.g., anticipated transients without scram, pressurized thermal shock, and operator guidelines) and is now available for one of the last remaining major issues still treated in a prescriptive manner, the loss-of-coolant accident.

The NRC staff amended § 50.46 of 10 CFR Part 50 to allow realistic methods to be used for the ECCS performance calculations in place of the evaluation models that use the required Appendix K features. This rule change also requires analysis of the uncertainty of the best-estimate calculations and requires that this uncertainty be considered when comparing the results of the calculations to the limits of paragraph 50.46(b) so that there is a high probability that

[6]Information Report from William J. Dircks to the Commissioners, dated November 17, 1983, "Emergency Core Cooling System Analysis Methods," SECY-83-472. Available for inspection or copying for a fee in the NRC Public Document Room, 2120 L Street NW., Washington, DC.

the criteria will not be exceeded. In this manner, more realistic calculations are available for regulatory decisions, yet an appropriate degree of conservatism would be maintained.

Many of the methods and models needed for a best-estimate calculation are the same as those used previously for evaluation model analyses. Although licensees and applicants are well acquainted with them, explicit guidance on acceptable methods and models (based on NRC experience with its own best-estimate advanced codes such as TRAC-PWR, TRAC-BWR, RELAP5, COBRA, and FRAP) would be useful. Further, the NRC has not previously published acceptable methods for uncertainty analyses. Therefore, guidance on methods acceptable to the NRC staff for calculating ECCS performance and for estimating the uncertainty are provided in the following Regulatory Position.

C. REGULATORY POSITION

1. BEST-ESTIMATE CALCULATIONS

A best-estimate calculation uses modeling that attempts to realistically describe the physical processes occurring in a nuclear reactor. There is no unique approach to the extremely complex modeling of these processes. The NRC has developed and assessed several best-estimate advanced thermal-hydraulic transient codes. These include TRAC-PWR, TRAC-BWR, RELAP5, COBRA, and the FRAP series of codes (References 1 through 6). These codes reasonably predict the major phenomena observed over a broad range of thermal-hydraulic and fuel tests. Licensees and applicants may use, but are not limited to, these codes and the specific models within them to perform best-estimate calculations of emergency core cooling system (ECCS) performance. Since the NRC staff has not performed the plant-specific uncertainty analysis required by the revised § 50.46 of 10 CFR Part 50, the licensee must demonstrate that the code and models used are acceptable and applicable to the specific facility over the intended operating range and must quantify the uncertainty in the specific application. General attributes expected in a best-estimate calculation are described here in Regulatory Position 1; special considerations for thermal-hydraulic best-estimate codes are presented in Regulatory Position 2; and specific examples of features that are considered acceptable best-estimate models are given in Regulatory Position 3. Other models or correlations will be considered acceptable if their technical basis is demonstrated with appropriate data and analysis.

A best-estimate model should provide a realistic calculation of the important parameters associated with a particular phenomenon to the degree practical with the currently available data and knowledge of the phenomenon. The model should be compared with applicable experimental data and should predict the mean of the data, rather than providing a bound to the data. The effects of all important variables should be considered. If it is not possible or practical to consider a particular phenomenon, the effect of ignoring this phenomenon should not normally be treated by including a bias in the analysis directly, but should be included as part of the model uncertainty. The importance of neglecting a particular phenomenon should be considered within the overall calculational uncertainty.

Careful consideration should be given to the range of applicability of a model when used in a best-estimate code. When comparing the model to data, judgments on the applicability of the data to the situation that would actually occur in a reactor should be made. Correlations generally should not be extrapolated beyond the range over which they were developed or assessed. If the model is to be extrapolated beyond the conditions for which valid data comparisons have been made, judgments should be made as to the effect of this extrapolation and the effect should be accounted for in the uncertainty evaluation. The fundamental laws of physics, well-established data bases (e.g., steam tables), and sensitivity studies should be used to assist in estimating the uncertainty that results from extrapolation.

A best-estimate code contains all the models necessary to predict the important phenomena that might occur during a loss-of-coolant accident. Best-estimate code calculations should be compared with applicable experimental data (e.g., separate-effects tests and integral simulations of loss-of-coolant accidents) to determine the overall uncertainty and biases of the calculation. In addition to providing input to the uncertainty evaluation, integral simulation data comparisons should be used to ensure that important phenomena that are expected to occur during a loss-of-coolant accident are adequately predicted. This is an idealized characterization of a best-estimate code. In practice, best-estimate codes may contain certain models that are simplified or that contain conservatism to some degree. This conservatism may be introduced for the following reasons:

1. The model simplification or conservatism has little effect on the result, and therefore the development of a better model is not justified.
2. The uncertainty of a particular model is difficult to determine, and only an upper bound can be determined.
3. The particular application does not require a totally best-estimate calculation, so a bias in the calculation is acceptable.

The introduction of conservative bias or simplification in otherwise best-estimate codes should not,

however, result in calculations that are unrealistic, that do not include important phenomena, or that contain bias and uncertainty that cannot be bounded. Therefore, any calculational procedure determined to be a best-estimate code in the context of this guide or for use under paragraph 50.46(a)(i) should be compared with applicable experimental data to ensure that the calculation of important phenomena is realistic.

2. CONSIDERATIONS FOR THERMAL-HYDRAULIC BEST-ESTIMATE CODES

Some features that are acceptable for use in best-estimate codes are described in the following paragraphs. Models that address these features may be used with the basic proviso that a specific model is acceptable if it has been compared with applicable experimental data and shown to provide reasonable predictions. Reference 7, "Compendium of ECCS Research for Realistic LOCA Analysis" (NUREG-1230), provides a summary of the large experimental data base available, upon which best-estimate models may be based. While inclusion in Reference 7 does not guarantee that the data or model will be acceptable, the report describes and references a large body of data generally applicable to best-estimate models. NUREG-1230 also provides documentation of NRC studies of the effect of reactor power increase on risk, background information on the ECCS rule, and a description of the methodology developed by NRC for estimating thermal-hydraulic transient code uncertainty.

For any models or correlations used in a best-estimate code, sufficient justification must be provided to substantiate that the code performs adequately for the classes of transients to which it applied. In general, the features of best-estimate thermal-hydraulic transient codes have uncertainties associated with their use for predicting reactor system response. These uncertainties should be considered as part of the overall uncertainty analysis described in Regulatory Position 4.

2.1 Basic Structure of Codes

2.1.1 Numerical Methods

A best-estimate code uses a numerical scheme for solving the equations used to predict the thermal-hydraulic behavior of the reactor. The numerical scheme is, in itself, a complex process that can play an important role in the overall calculation. Careful numerical modeling, sensitivity studies, and evaluations of numerical error should be performed to ensure that the results of the calculations are representative of the models used in the code. Numerical simulations of complex problems, such as those considered here, treat the geometry of the reactor in an approximate manner, making use of discrete volumes or nodes to represent the system. Sensitivity studies and evaluations of the uncertainty introduced by noding should be performed. Numerical methods treat time in a discrete manner, and the effect of time-step size should also be investigated.

2.1.2 Computational Models

A best-estimate code typically contains equations for conservation of mass, energy, and momentum of the reactor coolant and noncondensible gases, if important (e.g., air, nitrogen). Energy equations are also used to calculate the temperature distribution in reactor system structures and in the fuel rods. The required complexity of these equations will vary depending on the phenomena that are to be calculated and the required accuracy of the calculation. NRC staff experience with its own best-estimate computer codes has indicated that separate flow fields for different fluid phases, or types, and calculation of nonequilibrium between phases may be required to calculate some important phenomena (e.g., countercurrent flow, reflood heat transfer) to an acceptable accuracy. The NRC staff has also determined that certain phenomena require that the equations be solved in multiple dimensions. However, one-dimensional approximations to three-dimensional phenomena will be considered acceptable if those approximations are properly justified. Other basic code features include equations of state and other material properties. Sensitivity studies and comparisons to data should be performed to determine the importance of the simplifications used.

3. BEST-ESTIMATE CODE FEATURES

3.1 Initial and Boundary Conditions and Equipment Availability

The heat generated by the fuel during a loss-of-coolant accident depends on the power level of the reactor at the time of the loss-of-coolant accident and on the history of operation. The most limiting initial conditions expected over the life of the plant should be based on sensitivity studies. It is not necessary to assume initial conditions that could not occur in combination. For example, beginning-of-life peaking factors together with end-of-life decay heat do not require consideration. Given the assumed initial conditions, relevant factors such as the actual total power, actual peaking factors, and actual fuel conditions should be calculated in a best-estimate manner.

The calculations performed should be representative of the spectrum of possible break sizes from the full double-ended break of the largest pipe to a size small enough that it can be shown that smaller breaks are of less consequence than those already considered. The analyses should also include the effects of longitudinal splits in the largest pipes, with the split area equal to twice the cross-sectional area of the pipe. The range of break sizes considered should be

sufficiently broad that the system response as a function of break size is well enough defined so that interpolations between calculations, without considering unexpected behavior between the break sizes, may be made confidently.

Other boundary and initial conditions and equipment availability should be based on plant technical specification limits. These other conditions include, but may not be limited to, availability and performance of equipment, automatic controls, and operator actions. Appendix A to 10 CFR Part 50 requires that a single failure be considered when analyzing safety system performance and that the analysis consider the effect of using only onsite power and only offsite power.

3.2 Sources of Heat During a Loss-of-Coolant Accident

Models should account for the sources of heat discussed below and the distribution of heat production.

3.2.1 Initial Stored Energy of the Fuel

The steady-state temperature distribution and stored energy in the fuel before the postulated accident should be calculated in a best-estimate manner for the assumed initial conditions, fuel conditions, and operating history. To accomplish this, the thermal conductivity of the fuel pellets and the thermal conductance of the gap between the fuel pellet and the cladding should be evaluated. Thermal conductivity of fuel is a function of temperature and is degraded by the presence of gases in crack voids between fuel fragments. An acceptable model for thermal conductivity should be developed from the in-pile test results for fuel centerline and off-center temperatures, taking into account the conductivity of gases in crack voids.

Thermal conductance of the fuel-cladding gap is a strong function of hot gap size and of the composition and pressure of the gases in the fuel rod. The calculation of hot gap size should take into account UO_2 or mixed-oxide fuel swelling, densification, creep, thermal expansion and fragment relocation, and cladding creep. Fuel swelling is a function of temperature and burnup. Fuel densification is a function of burnup, temperature, and initial density. Densification can result from hydrostatic stresses imposed on fuel during pellet-cladding mechanical interaction and should be considered. Fuel creep is a function of time, temperature, grain size, density, fission rate, oxygen-to-metal ratio, and external stress. Fuel thermal expansion represents dimensional changes in unirradiated fuel pellets caused by changes in temperature. An acceptable model for the above fuel parameters should be based on in-pile and out-of-pile test data. Cladding creep introduces compressive creep strain in cladding during steady-state operation, reducing the gap between the fuel pellet and cladding. Cladding creep is a function of fast neutron flux (>1 MeV), cladding temperature, hoop stress, and material. Cladding materials may be cold-worked and stress-relieved or fully recrystallized, and there is a significant difference in the magnitude of creepdown between these materials. During pellet-cladding mechanical interaction, cladding experiences deformation from tensile creep, which is significantly different from that caused by compressive creep. An acceptable model for cladding tensile creep should be based on in-reactor tensile creep data.

Best-estimate fuel models will be considered acceptable provided the models include essential phenomena identified above and provided their technical basis is demonstrated with appropriate data and analyses.

3.2.1.1 Model Evaluation Procedure for Stored Energy and Heat Transfer in Fuel Rods. A model to be used in ECCS evaluations to calculate internal fuel rod heat transfer should:

a. Be checked against several sets of relevant data, and

b. Recognize the effects of fuel burnup, fuel pellet cracking and relocation, cladding creep, and gas mixture conductivity.

The model described by Lanning (Ref. 8) compared well to in-pile fuel temperature data. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.2.1.2 Experimental Data for Stored Energy in Fuel Rods and Heat Transfer. The correlations and data of Reference 9 are acceptable for calculating the initial stored energy of the fuel and subsequent heat transfer.

3.2.2 Fission Heat

Fission heat should be included in the calculation and should be calculated using best-estimate reactivity and reactor kinetics calculations. Shutdown reactivities resulting from temperatures and voids should also be calculated in a best-estimate manner. The point kinetics formulation is considered an acceptable best-estimate method for determining fission heat in safety calculations for loss-of-coolant accidents. Other best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. Control rod assembly insertion may be assumed if it is expected to occur.

3.2.3 Decay of Actinides

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 and known radioactive properties. The actinide decay heat chosen should be appropriate for the facility's operating history. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.2.4 Fission Product Decay Heat

The heat generation rates from radioactive decay of fission products, including the effects of neutron capture, should be included in the calculation and should be calculated in a best-estimate manner. The energy release per fission (Q value) should also be calculated in a best-estimate manner. Best-estimate methods will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. The model in Reference 10 is considered acceptable for calculating fission product decay heat.

3.2.4.1 Model Evaluation Procedure for Fission Product Decay Heat. The values of mean energy per fission (Q) and the models for actinide decay heat should be checked against a set of relevant data.

3.2.5 Metal-Water Reaction Rate

The rate of energy release, hydrogen generation, and cladding oxidation from the reaction of the zircaloy cladding with steam should be calculated in a best-estimate manner. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. For rods calculated to rupture their cladding during the loss-of-coolant accident, the oxidation of the inside of the cladding should be calculated in a best-estimate manner.

3.2.5.1 Model Evaluation Procedure for Metal-Water Reaction Rate. Correlations to be used to calculate metal-water reaction rates at less than or equal to 1900°F should:

a. Be checked against a set of relevant data, and

b. Recognize the effects of steam pressure, preoxidation of the cladding, deformation during oxidation, and internal oxidation from both steam and UO_2 fuel.

The data of Reference 11 are considered acceptable for calculating the rates of energy release, hydrogen generation, and cladding oxidation for cladding temperatures greater than 1900°F.

3.2.6 Heat Transfer from Reactor Internals

Heat transfer from piping, vessel walls, and internal hardware should be included in the calculation and should be calculated in a best-estimate manner. Heat transfer to channel boxes, control rods, guide tubes, and other in-core hardware should also be considered. Models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.2.7 Primary to Secondary Heat Transfer (Not Applicable to Boiling Water Reactors)

Heat transferred between the primary and secondary systems through the steam generators should be considered in the calculation and should be calculated in a best-estimate manner. Models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.3 Reactor Core Thermal/Physical Parameters

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

A calculation of the swelling and rupture of the cladding 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, should be included in the analysis and should be performed in a best-estimate manner. 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 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. Best-estimate methods to calculate the swelling of the 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. Best-estimate methods will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.3.2 Other Core Thermal Parameters

As necessary and appropriate, physical and chemical changes in in-core materials (e.g., eutectic formation, phase change, or other phenomena caused by material interaction) should be accounted for in the reactor core thermal analysis. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.4 Blowdown Phenomena

3.4.1 Break Characteristics and Flow

In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible break sizes should be considered, as indicated in Regulatory Position 3.1. The discharge flow rate should be calculated with a critical flow rate model that considers the fluid conditions at the break location, upstream and downstream pressures, and break geometry. The critical flow model should be justified by comparison to applicable experimental data over a range of conditions for which the model is applied. The model should be a best-estimate calculation, with uncertainty in the critical flow rate included as part of the uncertainty evaluation. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.4.1.1 Model Evaluation Procedure for Discharge Flow Rate. Critical flow models to be employed in ECCS evaluations should:

a. Be checked against an acceptable set of relevant data,

b. Recognize thermal nonequilibrium conditions when the fluid is subcooled, and

c. Provide a means of transition from nonequilibrium to equilibrium conditions.

The uncertainties and bias of a correlation or model used to calculate critical flow should be stated, as well as their range of applicability.

The mechanistic thermal nonequilibrium and slip model of Richter (Ref. 12) compares well to small- and large-scale test data (Ref. 13).

3.4.1.2 Experimental Data for Discharge Flow Rate. An acceptable set of relevant critical flow data should cover the fluid conditions, geometries, and types of breaks pertinent to light-water reactor loss-of-coolant accidents. The following tests should be considered in establishing an acceptable set of relevant data:

- Marviken tests (Ref. 14)
- Moby Dick experiments (Ref. 15)
- Brookhaven critical flashing flows in nozzles (Ref. 16)
- Sozzi-Sutherland tests (Ref. 17)
- Edwards experiments (Ref. 18)
- Super Moby Dick experiments (Refs. 19 and 20)

For critical flow from small breaks under stratified conditions, currently acceptable test data for assessing models and codes include those reported by:

- Anderson and Owca (Ref. 21)
- Reimann and Khan (Ref. 22)
- Schrock et al. (Refs. 23 and 24)

3.4.2 ECC Bypass

The best-estimate code should contain a calculation of the amount of injected cooling water that bypasses the vessel during the blowdown phase of the loss-of-coolant accident. The calculation of ECC bypass should be a best-estimate calculation using analyses and comparisons with applicable experimental data. Although it is clear that the dominant processes governing ECC bypass are multidimensional, single-dimensional approximations justified through sufficient analysis and data may be acceptable. Best-estimate methods will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. Cooling water that is not expelled, but remains in piping or is stored in parts of the vessel, should be calculated in a best-estimate manner based on applicable experimental data.

3.4.2.1 Model Evaluation Procedure for ECC Bypass. A correlation or model to be used to evaluate ECC bypass should:

a. Be checked against an acceptable set of relevant data, and

b. Recognize the effects of pressure, liquid subcooling, fluid conditions, hot walls, and system geometry.

Uncertainties and bias in the correlations or models used to calculate ECC bypass should be stated, as well as the range of their applicability.

For scaled-down PWR downcomers, correlations by Beckner and Reyes (Ref. 25) compared well to the bypass data of References 26 and 27. Correlations of Sun (Ref. 28) and Jones (Ref. 29) compare well to counter-current flow limiting (CCFL) test data of interest to BWRs.

3.4.2.2 Experimental Data for ECC Bypass. The following tests should be considered in establishing a set of data for scaled-down PWR downcomers:

- Battelle Columbus test (Ref. 26)
- Creare test (Refs. 26 and 27)

For a full-scale PWR vessel, ECC bypass data will become available from the forthcoming upper plenum test facility (UPTF) experiments performed as part of the 2D/3D program sponsored by the

Federal Republic of Germany, Japan, and the United States.

For BWRs, the following test should be considered in establishing an acceptable set of relevant data:

- SSTF test data (Refs. 30 through 32)

3.5 Noding Near the Break and ECCS Injection Point

The break location and ECCS injection point are areas of high fluid velocity and complex fluid flow and contain phenomena that are often difficult to calculate. The results of these calculations are often highly dependent on the noding. Sufficient sensitivity studies should be performed on the noding and other important parameters to ensure that the calculations provide realistic results.

3.6 Frictional Pressure Drop

The frictional losses in pipes and other components should be calculated using models that include variation of friction factor with Reynolds number and account for two-phase flow effects on friction. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.6.1 Model Evaluation Procedure for Frictional Pressure Drop

A model for frictional pressure drop to be used in ECCS evaluation should:

a. Be checked against a set of relevant data, and

b. Be consistent with models used for calculating gravitational and acceleration pressure drops. If void fraction models or correlations used to calculate the three components of the total pressure drop differ one from another, a quantitative justification must be provided.

Uncertainties and bias of a correlation or model should be stated as well as the range of applicability.

3.6.2 Experimental Data for Frictional Pressure Drop

An acceptable set of relevant data should cover, as far as possible, the ranges of parameters (mass flux, quality, pressure, fluid physical properties, roughness, and geometries) that are found in actual plant applications. The following tests should be considered in establishing an acceptable set of relevant data:

- Vertical tubes
  CISE test (Refs. 33 and 34)

- Horizontal tubes
  GE tests (Refs. 35 and 36)

- Rod bundles
  GE tests (Ref. 37)

3.7 Momentum Equation

The following effects should be taken into account in the two-phase conservation of momentum equation: (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. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.8 Critical Heat Flux

Best-estimate models developed from appropriate steady-state or transient experimental data should be used in calculating critical heat flux (CHF) during loss-of-coolant accidents. The codes in which these models are used should contain suitable checks to ensure that the range of conditions over which these correlations are used are within those intended. Research has shown that CHF is highly dependent on the fuel rod geometry, local heat flux, and fluid conditions. After CHF is predicted at an axial fuel rod location, the calculation may use nucleate boiling heat transfer correlations if the calculated local fluid and surface conditions justify the reestablishment of nucleate boiling. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.9 Post-CHF Blowdown Heat Transfer

Models of heat transfer from the fuel to the surrounding fluid in the post-CHF regimes of transition and film boiling should be best-estimate models based on comparison to applicable steady-state or transient data. Any model should be evaluated to demonstrate that it provides acceptable results over the applicable ranges. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.9.1 Model Evaluation Procedure for Post-CHF Heat Transfer

A model to be used in ECCS evaluation to calculate post-CHF heat transfer from rod bundles should:

a. Be checked against an acceptable set of relevant data, and

b. Recognize effects of liquid entrainment, thermal radiation, thermal nonequilibrium, low and high mass flow rates, low and high power

densities, and saturated and subcooled inlet conditions.

The uncertainties and bias of models or correlations used to calculate post-CHF heat transfer should be stated as well as the range of their applicability.

3.9.2 Experimental Data for Post-CHF Heat Transfer

The acceptable set of relevant data should cover power densities, mass flow rates, fluid conditions, and rod bundle geometries pertinent to light-water reactor designs and applications. The following tests should be considered in establishing an acceptable set of relevant data:

- ORNL tests (Refs. 38 and 39)
- FLECHT-SEASET tests (Ref. 40)
- INEL tests (Ref. 41)
- ORNL data base (Ref. 42)

3.9.3 Post-CHF Heat Transfer from Uncovered Bundles

During some time periods of small-break loss-of-coolant accidents and during portions of large breaks prior to reflood, partial or complete core uncovering may be calculated to occur. Under these circumstances, special considerations for calculating heat transfer are necessary.

3.9.3.1 Model Evaluation Procedures for Heat Transfer from Uncovered Rod Bundles. A correlation to be used in ECCS evaluations to calculate heat transfer from uncovered rod bundles should:

a. Be checked against an acceptable set of relevant data, and

b. Recognize the effects of radiation and of laminar, transition, and turbulent flows.

Uncertainties and bias in the models and correlations used to calculate post-CHF heat transfer should be stated, as should the range of their applicability.

The correlation derived should include a stated procedure for correcting for radiative heat transfer and for estimating the vapor temperatures. The Hottel procedure cited in Reference 43 is a satisfactory example.

The turbulent correlation may be of the general form:

$$Nu = A\,Re^{m}\,Pr^{n}$$

for higher Reynolds numbers (Re), where the coefficients A, m, and n are modifications from the basic Dittus-Boelter form and may be functions of other variables. Pr represents the Prandtl number, and Nu is the Nusselt number. The physical properties may be defined as wall, film, or vapor values.

A distinction from, and transition to, laminar convection (i.e., Re <2000) should be made, with a value of the laminar heat transfer for rod bundles that is appropriate for the applicable bundle geometry and flow conditions.

Other forms and values, depending on the bundle geometry and flow conditions, are also appropriate.

3.9.3.2 Experimental Data for Heat Transfer from Uncovered Rod Bundles. An acceptable set of relevant data for post-CHF heat transfer from uncovered rod bundles should cover power densities, fluid conditions, and rod bundle geometries pertinent to light-water reactor design and application. The following tests should be considered in establishing an acceptable set of relevant data:

- ORNL-THTF tests (Refs. 43 and 44)
- FLECHT-SEASET tests (Refs. 45 and 46)
- ORNL data base (Ref. 42)

3.10 Pump Modeling

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 pump speed as a function of time. The pump model resistance and other empirical terms should be justified through comparisons with applicable data. The pump model for the two-phase region should be verified by comparison to applicable two-phase performance data. Pump coastdown following loss of power should be treated in a best-estimate manner. A locked rotor following a large-break loss-of-coolant accident need not be assumed unless it is calculated to occur. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.11 Core Flow Distribution During Blowdown

The core flow through the hottest region of the core during the blowdown should be calculated as a function of time. For the purpose of these calculations, the hottest region of the core should not be greater than the size of one fuel assembly. Calculations of the flow in the hot region should take into account any cross-flow between regions and any flow blockage calculated to occur during the blowdown as a result of cladding swelling or rupture. The numerical scheme should ensure that unrealistic oscillations of the calculated flow do not result. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.12 Post-Blowdown Phenomena

3.12.1 Containment Pressure

The containment pressure used for evaluating cooling effectiveness during the post-blowdown phase of a loss-of-coolant accident should be calculated in a best-estimate manner and should include the effects of containment heat sinks. The calculation should include the effects of operation of all pressure-reducing equipment assumed to be available. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

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

The refilling of the reactor vessel and the ultimate reflooding of the core should be calculated by a best-estimate model that takes into consideration the thermal and hydraulic characteristics of the core, the emergency core cooling systems, and the primary and secondary reactor systems. The model should be capable of calculating the two-phase level in the reactor during the postulated transient. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.12.2.1 Model Evaluation Procedures for Post-Blowdown Thermal Hydraulics. A correlation or model to be used in ECCS evaluation to calculate level swell should be checked against an acceptable set of relevant data and should recognize the effects of depressurization, boil-off, power level, fluid conditions, and system geometry.

The correlation proposed by Chexal, Horowitz, and Lellouche (Ref. 47) provides acceptable results when compared to experimental data reported in References 43, 48, 49, and 50.

Uncertainties and bias of a correlation or model used to calculate level swell should be stated, as should the range of applicability.

The primary coolant pumps should be assumed to be operating in the expected manner, based on the assumptions of Regulatory Position 3.1, when calculating the resistance offered by the pumps to fluid flow. Models will be considered acceptable provided their technical basis is demonstrated through comparison with appropriate data and analyses.

The total fluid flow leaving the core exit (carryover) should be calculated using a best-estimate model that includes the effect of cross-flow on carryover and core fluid distribution. Thermal-hydraulic phenomena associated with unique emergency core cooling systems, such as upper plenum injection and upper head injection, should be accounted for. The effects of the compressed gas in the accumulator following accumulator water discharge should be included in the calculation. Any model or code used for this calculation should be assessed against applicable experimental data. Reference 7 describes a large body of refill/reflood thermal-hydraulic data obtained from the 2D/3D program that is appropriate for consideration.

3.12.2.2 Experimental Data for Post-Blowdown Thermal Hydraulics. The following tests should be considered when establishing an acceptable set of relevant data:

- GE tests (Refs. 48 and 51)
- ORNL tests (Refs. 43 and 49)
- FLECHT-SEASET test (Ref. 45)
- THETIS tests (Ref. 50)

3.12.3 Steam Interaction with Emergency Core Cooling Water in Pressurized Water Reactors

The thermal-hydraulic interaction between the steam or two-phase fluid and the emergency core cooling water should be taken into account in calculating the core thermal hydraulics and the steam flow through the reactor coolant pipes during the time the accumulators are discharging water. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.

3.12.4 Post-Blowdown Heat Transfer for Pressurized Water Reactors

During refilling of the reactor vessel and ultimate reflooding of the core, the heat transfer calculations should be based on a best-estimate calculation of the fluid flow through the core, accounting for unique emergency core cooling systems. The calculations should also include the effects of any flow blockage calculated to occur as a result of cladding swelling or rupture. Heat transfer calculations that account for two-phase conditions in the core during refilling of the reactor vessel should be justified through comparisons with experimental data. Best-estimate models will be considered acceptable provided their technical basis is demonstrated through comparison with appropriate data and analyses.

The FLECHT-SEASET tests (Refs. 40, 45, and 46) should be considered when establishing an acceptable set of relevant data. Reference 7 contains extensive information regarding a large amount of experimental reflood heat transfer data. This information should also be considered when developing and assessing models. The results from the 2D/3D program are particularly relevant.

3.13 Convective Heat Transfer Coefficients for Boiling Water Reactor Rods Under Spray Cooling

Models will be considered acceptable provided their technical bases can be justified with appropriate data and analyses. These models should contain the following:

1. Following the blowdown period, convective heat transfer coefficients should be determined based on the calculated fluid conditions and heat transfer modes within the bundle and on the calculated rod temperatures.

2. During the period following the flashing of the lower plenum fluid, but prior to ECCS initiation, heat transfer models should include cooling by steam flow or by a two-phase mixture, if calculated to occur.

3. Following initiation of ECCS flow, but prior to reflooding, heat transfer should be based on the actual calculated bundle fluid conditions and best-estimate heat transfer models that take into account rod-to-rod variations in heat transfer.

4. After the two-phase reflood 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 calculated to occur as a result of cladding swelling or rupture.

5. Thermal-hydraulic models that do not calculate multiple channel effects should be compared with applicable experimental data or more detailed calculations to ensure that all important phenomena are adequately calculated.

3.14 Boiling Water Reactor Channel Box Under Spray Cooling

Following the blowdown period, heat transfer from the channel box and wetting of the channel box should be based on the calculated fluid conditions on both sides of the channel box and should make use of best-estimate heat transfer and rewetting models that have been compared with applicable experimental data.

3.15 Special Considerations for a Small-Break Loss-of-Coolant Accident in Pressurized Water Reactors

The slower small-break loss-of-coolant accident leads to fluid conditions characterized by separation of the fluid phases versus the more homogeneous fluid conditions that would result from rapid large-break loss-of-coolant accident transients. Phenomena that would occur in a PWR during a small-break loss-of-coolant accident would, therefore, be significantly different from those phenomena that would occur during a large-break loss-of-coolant accident. The distribution of liquid throughout the reactor system, in addition to the total liquid inventory, is of increased importance for the small-break loss-of-coolant accident. A number of special factors must be given increased consideration in small-break loss-of-coolant accident calculations to correctly predict phenomena influenced by the liquid inventory distribution.

Break flow may be greatly influenced by the location and specific geometry of the break. For a break in a horizontal pipe containing stratified flow, the quality of the break flow will be a strong function of the assumed location of the break on the pipe (e.g., top or bottom). Small-break loss-of-coolant accident calculations should, therefore, include various assumed break locations in the spectrum of breaks analyzed. The assumed operating state of the reactor coolant pump will also influence the distribution of liquid throughout the system and the amount of liquid lost through the break.

The pump operation assumptions used in the calculations should be the most likely, based on operating procedures, with appropriate consideration of the uncertainty of the pump operation during an actual event. Level depression in the core region and subsequent core heatup may be influenced by liquid holdup in the steam generator tubes, manometric effects of liquid in the piping and loop seal region, and liquid levels relative to vent paths for steam through upper plenum bypass flow paths and vent valves. Steam generator heat transfer under "reflux" or "boiler-condensor" modes of operation may also strongly influence core inventory through level depression and the effect on total system pressure and, thus, on ECCS flow. These phenomena should be carefully considered in the calculation. Sensitivity studies of the importance of these effects should be performed for use in the uncertainty evaluation.

Heat transfer from an uncovered core under high-pressure conditions typical during a small-break loss-of-coolant accident may include contributions from both convective and radiation heat transfer to the steam. Models will be considered acceptable provided their technical basis is demonstrated through comparison with appropriate data and analyses. Specific guidance regarding uncovered bundle heat transfer is given in Regulatory Position 3.9.3.

3.16 Other Features of Best-Estimate Codes

No list of best-estimate code features could be all-inclusive, because the important features of a best-estimate code may vary depending on the transient to be calculated and the required accuracy of the calculation. Because of this, no attempt has been made to construct an exhaustive list of best-estimate code features. Rather, features that were identified as

important for inclusion in Appendix K were used as a basis for the above list. These features are not necessarily any more or less important than other code features, but were highlighted because it is necessary to give specific examples of how current best-estimate models may vary from methods used traditionally in evaluation model codes using the various Appendix K conservatisms. In addition, models have not been included for areas in which the best model would be highly dependent on the specific plant design or the specific transient under consideration.

The NRC staff believes that good examples of best-estimate thermal-hydraulic transient codes are those developed by the NRC (e.g., TRAC-PWR, TRAC-BWR, RELAP5, COBRA, and FRAP). Although these codes are subject to further improvement, based on their ongoing use and assessment, they currently provide reasonable best-estimate calculations of the loss-of-coolant accident in a full-scale light-water reactor. This is substantiated through the code development and assessment literature generated by the NRC and its contractors over the past several years.

It is possible, however, to describe in general how other features of best-estimate codes should be constructed. Two basic criteria should be applied, completeness and comparisons to experimental data.

3.16.1 Completeness

Best-estimate codes should contain models in sufficient detail to predict phenomena that are important to demonstrate compliance with the acceptance criteria specified in paragraph 50.46(b) of 10 CFR Part 50 (e.g., peak cladding temperature). Simplifications are acceptable as long as code uncertainties or biases do not become so large that they cast doubt on the actual behavior that would occur or on the true effect of assumed initial and boundary conditions (e.g., equipment sizing, safety system settings). Comparisons of the overall calculations to integral experiments should be performed to ensure that important phenomena can be predicted and to help in making judgments on the effect of code simplifications. Consideration should also be given to the uncertainty and validity of the experiment to ensure that meaningful comparisons are being made.

3.16.2 Data Comparisons

Individual best-estimate models should be compared to applicable experimental data to ensure that realistic behavior is predicted and that relevant experimental variables are included. Uncertainty analyses are required to ensure that a major bias does not exist in the models and that the model uncertainty is small enough to provide a realistic estimate of the effect of important experimental variables. Uncertainty analyses should also consider experimental uncertainty to ensure that meaningful comparisons are being made.

4. ESTIMATION OF OVERALL CALCULATIONAL UNCERTAINTY

4.1 General

The term "uncertainty," when applied to best-estimate thermal-hydraulic transient codes, is used at two levels. At the lower or more detailed level, the term refers to the degree to which an individual model, correlation, or method used within the code represents the physical phenomenon it addresses. These individual uncertainties, when taken together, comprise the "code uncertainty."

The combined uncertainty associated with individual models (i.e., code uncertainty) within the best-estimate codes does not account for all of the uncertainty associated with the model's use. In addition to the code uncertainty, various other sources of uncertainty are introduced when attempting to use best-estimate codes to predict full-scale plant thermal-hydraulic response. These sources include uncertainty associated with the experimental data used in the code assessment process (including applicability of the data to full-scale reactors), the input boundary and initial conditions, and the fuel behavior. Additional sources of uncertainty stem from the use of simplifying assumptions and approximations. A careful statement of these assumptions and approximations should be made, and the uncertainty associated with them should be taken into account. Therefore, the "overall calculational uncertainty" is defined as the uncertainty arrived at when all the contributions from the sources identified above, including the code uncertainty, are taken into account.

A 95% probability level is considered acceptable to the NRC staff for comparison of best-estimate predictions to the applicable limits of paragraph 50.46(b) of 10 CFR Part 50 to meet the requirement of paragraph 50.46(a)(1)(i) to show that there is a high probability that the criteria will not be exceeded. The basis for selecting the 95% probability level is primarily for consistency with standard engineering practice in regulatory matters involving thermal hydraulics. Many parameters, most notably the departure from nucleate boiling ratio (DNBR), have been found acceptable by the NRC staff in the past at the 95% probability level.

This 95% probability level would also be applied to small-break loss-of-coolant accidents, which have a higher probability than large breaks. The dominant factors influencing risk from small-break loss-of-coolant accidents include equipment availability and operator actions. Calculational uncertainties are much less important than factors such as operator recognition of the event, the availability of equipment, and the correct use of this equipment. The use

of a best-estimate calculation with reasonable and quantifiable uncertainty is expected to provide a reduction in the overall risk from a small-break loss-of-coolant accident by providing more realistic calculations with which to evaluate operator guidelines and determine the true effect of equipment availability.

Regulatory Position 3 provides a description of the features that should be included in the overall code uncertainty evaluation that is called for in paragraph 50.46(a)(1). This uncertainty evaluation should make use of probabilistic and statistical methods to determine the code uncertainty. For a calculation of this complexity, a completely rigorous mathematical treatment is neither practical nor required. In many cases, approximations and assumptions may be made to make the overall calculational uncertainty evaluation possible. A careful statement of these assumptions and approximations should be made so that the NRC staff may make a judgment as to the validity of the uncertainty evaluation. The purpose of the uncertainty evaluation is to provide assurance that for postulated loss-of-coolant accidents a given plant will not, with a probability of 95% or more, exceed the applicable limits specified in paragraph 50.46(b).

4.2 Code Uncertainty

This regulatory guide makes a distinction between the terms "code uncertainty" and "overall calculational uncertainty." The latter term is defined above and includes the contributions to the uncertainty described in Regulatory Positions 4.2 and 4.3. The components of the code uncertainty (i.e., the contribution to the overall uncertainty from the models and numerical methods used) are described in this section.

Code uncertainty should be evaluated through direct data comparison with relevant integral systems and separate-effects experiments at different scales. In this manner, an estimate of the uncertainty attributable to the combined effect of the models and correlations within the code can be obtained for all scales and for different phenomena. Comparison to a sufficient number of integral systems experiments, from different test facilities and different scales, should be made to ensure that a reasonable estimate of code uncertainty and bias has been obtained. When necessary, separate-effects experiments should be used to establish code uncertainty for specific phenomena (e.g., comparisons to Cylindrical Core Test Facility data to ascertain code uncertainty in modeling upper plenum injection performance). Code comparisons should account for limitations of the measurements and calibration errors.

These comparisons should be performed for important key parameters to demonstrate the overall best-estimate capability of the code. For large-break loss-of-coolant accidents, the most important key parameter is peak cladding temperature, which is addressed by one of the criteria of paragraph 50.46(b) and has a direct influence on the other criteria. In addition, a code uncertainty evaluation should be performed for other important parameters for the transient of interest to evaluate compensating errors. For small-break loss-of-coolant accidents, the cladding temperature response is the most important parameter; however, the ability of the codes to predict overall system mass and reactor vessel inventory distribution should also be statistically examined.

In evaluating the code uncertainty, it will be necessary to evaluate the code's predictive ability over several time intervals, since different processes and phenomena occur at different intervals. For example, in large-break loss-of-coolant accident evaluations, separate code uncertainties may be required for the peak cladding temperature during the blowdown and post-blowdown phases. Justification for treating these uncertainties individually or methods for combining them should be provided.

The experimental information used to determine code uncertainty will usually be obtained from facilities that are much smaller than nuclear power reactors. Applicability of these results should be justified for larger scales. The effects of scale can be assessed through comparisons to available large-scale separate-effects tests and through comparison to integral tests from various sized facilities. If there are scaling problems, particularly if predictions are nonconservative, the code should be improved for large-scale plants (i.e., nuclear reactors). Codes not having scaling capability will not be acceptable if their predictions are nonconservative.

4.3 Other Sources of Uncertainty

When a best-estimate methodology is used to predict reactor transients, sources of uncertainty other than the limitations in the individual models and numerical methods (i.e., code uncertainty) are introduced. The following contributors to the overall calculational uncertainty should also be considered in the uncertainty analysis.

4.3.1 Initial and Boundary Conditions and Equipment Availability

When a plant input model is prepared, certain relationships describing the plant boundary and initial conditions and the availability and performance of equipment are defined. These include factors such as initial power level, pump performance, valve activation times, and control systems functioning. Uncertainties associated with the boundary and initial conditions and the characterization and performance of equipment should be accounted for in the uncertainty evaluation. It is also acceptable to limit the variables

to be considered by setting their values to conservative bounds.

4.3.2 Fuel Behavior

Variability of the results of plant transient calculations can result from uncertainties associated with fuel behavior, which are not included in the comparisons of code results with integral experiments since most integral tests use electrically heated rods. This uncertainty includes many effects such as fuel conductivity, gap width, gap conductivity, and peaking factors. These uncertainties should be quantified and used in the determination of the overall calculational uncertainty.

4.3.3 Other Variables

There may be individual models within the best-estimate code whose effect may not have been evaluated by the comparison to the integral systems data. For example, since most integral systems experiments use electrically heated rods, uncertainties associated with the prediction of core decay heat and cladding metal/water reaction have not been evaluated. In addition, to demonstrate the overall adequacy of the predictive ability of the best-estimate code, it may be necessary to use empirically arrived at break-discharge coefficients to obtain a reasonable break flow. The uncertainties in the individual models that have not been evaluated by comparison to integral systems data should be quantified and used in the determination of overall code uncertainty.

4.4 Statistical Treatment of Overall Calculational Uncertainty

The methodology used to obtain an estimate of the overall calculational uncertainty at the 95% probability limit should be provided and justified. If linear independence is assumed, suitable justification should be provided. The influence of the individual parameters on code uncertainty should be examined by making comparisons to relevant experimental data. Justification should be provided for the assumed distribution of the parameter and the range considered.

In reality, the true statistical distribution for the key parameters (e.g., peak cladding temperature) is unknown. The choice of a statistical distribution should be verified using applicable engineering data and information. The statistical parameters appropriate for that distribution should be estimated using available data and results of engineering analyses. Supporting documentation should be provided for this selection process. These estimated values are assumed to be the true values of the statistical parameters of the distribution. With these assumptions, an upper one-sided probability limit can be calculated at the 95% level. As the probability limit approaches 2200°F, more care must be taken in the selection and justification of the statistical distribution and in the estimation of its statistical parameters. If a normal distribution is selected and justified, the probability limit can be conservatively calculated using two standard deviations. The added conservatism of the two standard deviations compared to the 95th percentile is used to account for uncertainty in the probability distribution. Other techniques that account for the uncertainty in a more detailed manner may be used. These techniques may require the use of confidence levels, which are not required by the above approach.

The evaluation of the peak cladding temperature at the 95% probability level need only be performed for the worst-case break identified by the break spectrum analysis in order to demonstrate conformance with paragraph 50.46(b). However, in order to use this approach, justification must be provided that demonstrates that the overall calculational uncertainty for the worst case bounds the uncertainty for other breaks within the spectrum. It may be necessary to perform separate uncertainty evaluations for large- and small-break loss-of-coolant accidents because of the substantial difference in system thermal-hydraulic behavior.

The revised paragraph 50.46(a)(1)(i) requires that it be shown with a high probability that none of the criteria of paragraph 50.46(b) will be exceeded, and is not limited to the peak cladding temperature criterion. However, since the other criteria are strongly dependent on peak cladding temperature, explicit consideration of the probability of exceeding the other criteria may not be required if it can be demonstrated that meeting the temperature criterion at the 95% probability level ensures with an equal or greater probability that the other criteria will not be exceeded.

4.5 NRC Approach to LOCA Uncertainty Evaluation

Chapter 4 of the "Compendium of ECCS Research for Realistic LOCA Analysis" (Ref. 7) presents a methodology that has been used for evaluating the overall calculational uncertainty in peak cladding temperature predictions for best-estimate thermal-hydraulic transient codes that the NRC has developed.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Licensees and applicants may propose means other than those specified by the provisions of the Regulatory Position of this guide for meeting applicable regulations. This guide has been approved for use by the NRC staff as an acceptable means of

complying with the Commission's regulations and for evaluating submittals in the following categories:

1. Construction permit applicants that choose to make use of the provisions of § 50.46 that allow the use of realistic models as an alternative to the features of Appendix K of 10 CFR Part 50.

2. Operating license applicants that choose to make use of the provisions of § 50.46 that allow the use of realistic models as an alternative to the features of Appendix K of 10 CFR Part 50.

3. Operating reactor licensees will not be evaluated against the provisions of this guide except for new submittals that make use of the provisions of § 50.46 that allow the use of realistic models as an alternative to the features of Appendix K of 10 CFR Part 50.

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ADDRESSES

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REGULATORY ANALYSIS

A separate regulatory analysis has not been prepared in support of this regulatory guide. The regulatory analysis that supports the rulemaking effort also covers this regulatory guide. "Regulatory Analysis for Revision of the ECCS Rule and Supporting Regulatory Guide," is available in the NRC Public Document Room, 2120 L Street NW., Washington, DC, under Regulatory Guide 1.157 (52 FR 6334).

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