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PART 1: AN OVERVIEW OF THE CODE SCALING, APPLICABILITY, AND UNCERTAINTY EVALUATION METHODOLOGY

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QUANTIFYING REACTOR SAFETY MARGINS PART 1: AN OVERVIEW OF THE CODE SCALING, APPLICABILITY, AND UNCERTAINTY EVALUATION METHODOLOGY*

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ABSTRACT

In August 1988, the Nuclear Regulatory Commission (NRC) approved the final version of a revised rule on the acceptance of emergency core cooling systems (ECCS) entitled "Emergency Core Cooling System; Revisions to Acceptance Criteria." The revised rule states an alternate ECCS performance analysis, based on best-estimate methods, may be used to provide more realistic estimates of plant safety margins, provided the licensee quantifies the uncertainty of the estimates and includes that uncertainty when comparing the calculated results with prescribed acceptance limits.

To support the revised ECCS rule, the NRC and its contractors and consultants have developed and demonstrated a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology. It is an auditable, traceable, and practical method for combining quantitative analyses and expert opinions to arrive at computed values of uncertainty.

This paper provides an overview of the CSAU evaluation methodology and its application to a postulated cold-leg, large-break loss-of-coolant accident in a Westinghouse four-loop pressurized water reactor with 17 x 17 fuel. The code selected for this demonstration of the CSAU methodology was TRAC-PF1/MOD1, Version 14.3.

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I. INTRODUCTION

In August 1988, the Nuclear Regulatory Commission (NRC) approved the final version of a revised rule on the acceptance of emergency core cooling systems (ECCS) entitled "Emergency Core Cooling System; Revision to Acceptance Criteria" (Ref. 1). The revised rule contains three key features. First, the current acceptance criteria related to peak cladding temperature, clad oxidation, hydrogen generation, coolable core geometry, and long-term cooling are retained. Second, evaluation model (EM) methods based on Appendix K (Ref. 1) may continue to be used as an alternative to the best-estimate (BE) methodology. Third, an alternate ECCS performance analysis, based on BE methods, may be used to provide more realistic estimates of plant safety margins, provided the licensee quantifies the uncertainty of the estimates and includes that uncertainty when comparing the calculated results with prescribed acceptance limits (Ref. 2).

To support the revised ECCS rule, the NRC and its contractors and consultants have developed and demonstrated a method called the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology. The objective of this paper is to provide an overview of this methodology and a brief summary of its application to a large-break loss-of-coolant accident (LBLOCA). More detailed descriptions of specific features of the CSAU method and its demonstration for a Westinghouse four-loop pressurized water reactor (PWR) with 17 x 17 fuel are presented in companion papers (Refs. 3–5).

II. BACKGROUND

Recent review papers concerning the history and content of the ECCS rules are found in Refs. 6–8. The background material provided here was summarized from these papers. The acceptance criteria for ECCS performance for light-water-cooled nuclear power plants are found in the Code of Federal Regulations, Title 10, Section 50.46 (10CFR50.46) (Ref. 9). Included is the requirement that analysis models used to calculate the thermal-hydraulic performance of the ECCS conform to the requirements specified in Appendix K (Ref. 1) to 10CFR50. Section 50.46 and Appendix K were finalized after extensive public hearings in 1973, and the rule was implemented in January 1974. The basic criteria for evaluating ECCS performance focus on a peak cladding temperature (PCT) limit (2200°F or 1477 K), a limit on the maximum cladding oxidation (cannot exceed 17% of the cladding thickness before oxidation), a limit on the hydrogen generation from the chemical reaction of the cladding with water or steam (1% of potential), a requirement that a coolable core geometry be retained, and a requirement that acceptable long-term cooling be provided. These criteria were re-evaluated as part of the rule revision process and were found, in the judgement of the NRC, to be appropriate for continued use. Appendix K contains required and acceptable features of evaluation models to be used for loss of coolant accident (LOCA) analysis.

Several features of the Appendix K requirements have been found to have a significant effect on PWR design and operation. These are requirements related to initial stored energy, use of 1.2 times the 1971–73 American Nuclear Society standard decay heat, the emergency core coolant (ECC) bypass prescription, the prohibition on a return to nucleate boiling, reactor coolant pump modeling, and calculation of reflood rates (Ref. 10). Other features of Appendix K were found to have smaller impacts on plant design and operation. A summary of an early study (Ref. 11) to quantify the conservatisms associated with a particular set of

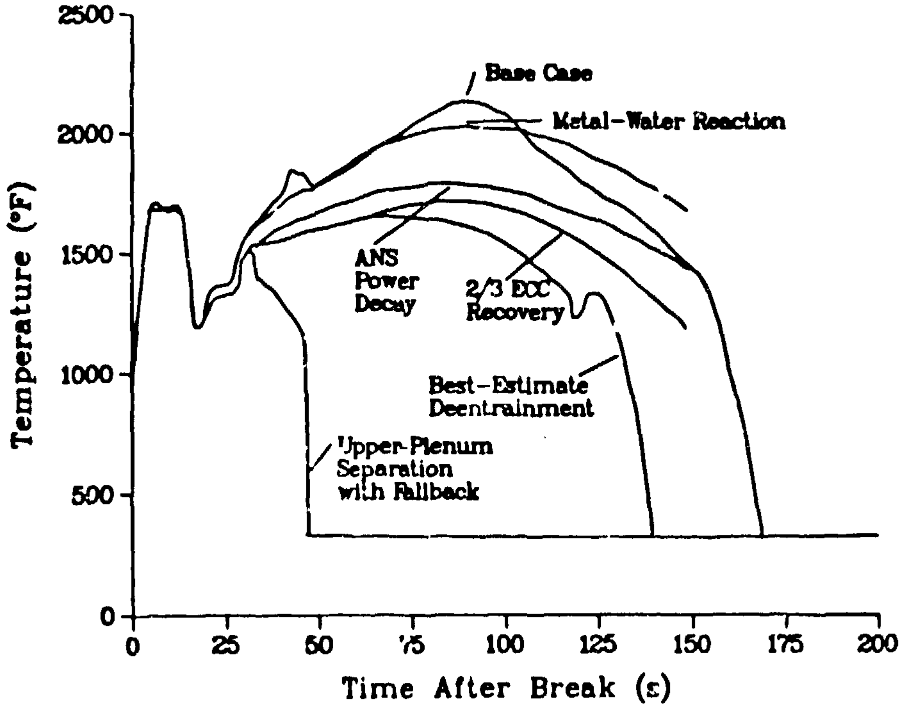

Fig. 1.
Effect of selected conservative assumptions on PCT.

Appendix K requirements is provided in Fig. 1. Following the base-case code calculation using EM conditions, selected conservatisms were removed one at a time and the transient recalculated. Taken in the order below, the conservatisms accounted for a reduction in the reflood PCT of about 700°F (390 K); the order in which the selected Appendix K conservatisms were treated was (1) metal-water reaction, (2) decay heat, (3) ECCS bypass, (4) upper-plenum de-entrainment, and (5) upper-plenum separation with fallback. The blowdown PCT was only slightly affected by treatment of the listed conservatisms.

There appear to be many areas of possible benefit available to licensees as increased margin is demonstrated under the revised ECCS rule. Possible benefits for PWR plants include increased discharge burnup, low leakage loading patterns, longer fuel cycles, use of advanced fuel designs, power uprating, improved load-following capabilities, more flexible burnup windows, use of axial blankets, and technical specification relaxation (Ref. 12). Similar insights related to potential benefits for boiling-water reactors (BWRs) are found in Ref. 13. In 1985, a BE licensing approach based on the SAFER/GESTR codes was licensed by the NRC for application to BWRs. This approach has subsequently been used to identify areas where this licensed technology could be used to relax ECCS performance requirements for BWR/4 (Ref. 14).

III. CSAU EVALUATION METHODOLOGY

A. Objectives of CSAU

The objectives of the CSAU evaluation methodology developed by NRC and its contractors and consultants are to:

1. provide a technical basis for quantifying uncertainties within the context of the revised ECCS rule;
2. provide an auditable, traceable and practical method for combining quantitative analyses and expert opinion to arrive at computed values of uncertainty; and
3. provide a systematic and comprehensive approach for:
   - defining scenario phenomena
   - evaluating code applicability
   - assessing code scale-up capabilities, and
   - quantifying code uncertainties concerned with:
     - code and experiment accuracies
     - code scale-up capabilities
     - plant state and operating conditions.

Additional objectives of the CSAU activities described in this and the companion papers were to "demonstrate feasibility of the methodology, to develop an audit tool, and to provide the necessary experience to audit licensee submittals" (Ref. 1).

B. Elements of CSAU

In developing CSAU, the emphasis was placed on providing a **practical engineering approach** that could be used to quantify code uncertainties. Consequently, for a specified nuclear power plant (NPP) and a given scenario, the CSAU method focuses only on important processes and/or phenomena, assesses the code capability to scale them up, and evaluates the accuracy with which the code calculates them.

The CSAU evaluation methodology consists of three primary elements as shown in Fig. 2.

- The first element, **Requirements and Code Capabilities**, contains steps 1–6. In this element, scenario modeling requirements are identified and compared against code capabilities to determine the code's applicability to the particular scenario in a given plant type. In addition, an effort is made to identify potential code limitations.

  The modeling requirements are established by identifying and ranking problems and phenomena important to the particular scenarios. The need for such a screening process arises from the fact that the resources required to quantify the uncertainty for **every** process and phenomenon occurring during the selected transient are **too large**. Furthermore, although many processes and phenomena occur during a given transient, the overall plant behavior is not influenced equally by each. Consequently, a sufficient and more efficient (cost effective) approach is to rank process and phenomena by evaluating their importance relative to the primary safety criteria so that **only** the significant contributors need to be evaluated. A list of the processes and phenomena occurring during the selected transient is compiled following examination of experimental data and code simulations.

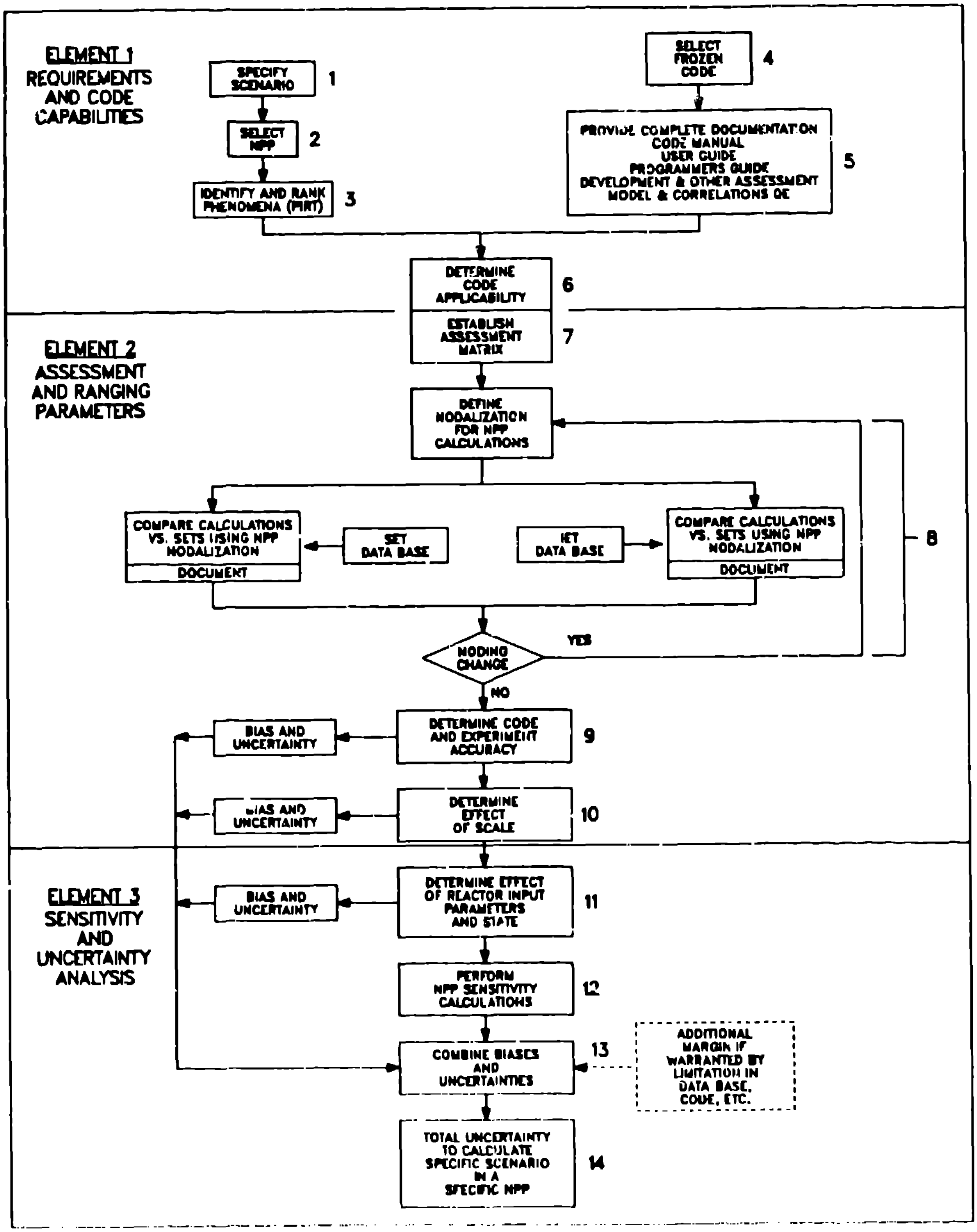

Fig. 2.
Code scaling, applicability, and uncertainty evaluation methodology.

The phenomena and processes are then ranked according to importance using one or more techniques.

The evaluation of code applicability to the selected transient is based on a complete set of code documents (specified in step 5). It is conducted by reviewing the adequacy of code models to calculate the important processes and phenomena identified above. Code deficiencies and/or limitations are also identified and evaluated as to their potential affects on uncertainties to calculate primary safety criteria. A more detailed description of this element and its demonstration for a LBLOCA is provided in Ref. 3.

- The second element, **Assessment and Ranging of Parameters**, contains steps 7–10. In this element there are activities to assess the capability of the code to calculate processes important to the scenario by comparing calculations against experimental data to determine code accuracy, to determine scale-up capability, and to specify ranges of parameter variations needed for sensitivity studies. In addition, bounding analyses can be performed and, in such cases, code calculations may not be required. A more detailed description of this element and its demonstration is provided in Ref. 4.
- The third element, **Sensitivity and Uncertainty Analyses**, contains steps 11–14. The total uncertainty in a safety analysis includes contributors that arise from code limitations, scaling inaccuracies embedded in the experimental data (and therefore the code), and uncertainties associated with the state of the reactors at the initiation of a transient. The ultimate objective of the CSAU process is to provide a simple and direct statement of the calculated uncertainty in the primary safety criteria (e.g., the PCT) used as the basis for assessing safety and making licensing decisions relative to the revised ECCS rule. This objective is accomplished when the magnitudes of individually important contributors are determined, collected, and subsequently combined to provide the desired summary statement. This element contains the activities to calculate, collect, and combine individual contributors to uncertainty into the required total mean and 95% probability statements including separately identified and quantified biases. A more detailed description of this element and its demonstration is provided in Ref. 5.

C. Prescriptive Steps of CSAU

A brief description of each of the steps (numbered to conform to Fig. 2) in the CSAU evaluation methodology follows.

1. **Specify Scenario**: Code applicability and uncertainty are transient dependent because processes and safety parameters of interest may change from one scenario to another. Consequently, it is necessary to specify the scenario to order to establish the parameters that need to evaluated.
2. **Select Nuclear Power Plant (NPP)**: The scenario definition depends on both the type of transient to be analyzed, and the particular plant in which it is postulated that it occurs. Consequently, the NPP for which the calculations are to be performed should be specified.

3. **Identify and Rank Processes**: The CSAU methodology focuses only on phenomena/processes that are important to the particular scenario and plant design. Consequently, physical processes need to be first identified (together with relevant plant components) and then ranked to establish process identification and ranking tables (PIRT) appropriate to the particular scenario and plant design. The identification and ranking should be justified and documented.
4. **Select "Frozen" Code**: The methodology emphasizes the use of a "frozen" code version. This ensures that changes to the code after an evaluation has been completed do not impact the conclusions, and that changes occur in an **auditable** and **traceable** manner.
5. **Provide Code Documentation**: Complete documentation should include: the user manual, the user guide, the model and correlations quality evaluation (MC/QE) document, assessment reports, and other relevant supporting evidence. The MC/QE is a new kind of document that NRC's contractors have been requested to provide for each BE code. It provides detailed information on closure relations as implemented in the code, that is, information on their source, data base, accuracy, scale-up capability, and applicability to NPP conditions.
6. **Determine Code Applicability**: Steps 1 through 3 establish the requirements for modeling a specific scenario, whereas steps 4 and 5 provide information on code capabilities. By comparing requirements and capabilities in step 5, one determines whether the code can be used to calculate the scenario and whether the data base, accuracy, and scale-up capabilities of closure relations are adequate to model processes important to the scenario.
7. **Establish an Assessment Matrix**: Test data should be selected for a) assessing code accuracy to calculate dominant phenomena/processes identified in PIRT (step 3) and b) addressing any code inadequacy identified in step 6. Both separate effects tests (SET) and integral effects tests (IET) are to be used in establishing the assessment matrix.
8. **Define NPP Nodalization**: The plant model should be nodalized with enough detail to capture the important phenomena and design characteristics of the NPP. The iterative process shown in the flow diagram (Fig. 2) makes use of the Assessment Matrix to support the rationale for the choice of nodalization. The same nodalization should be used in performing NPP and code assessment calculations.
9. **Determine Code and Experiment Accuracy**: Simulations of experiments from step 7 with the frozen code, using the nodalization defined above, should lead to a statement of code accuracy. Differences between code and experiment should be quantified for bias and uncertainty (deviation). Individual contributors to uncertainties in modeling important phenomena/processes should be identified and cast in terms of bias and distribution (to be used in step 12 for sensitivity calculations). In addition, separate biases should be evaluated as appropriate.

   SET data from facilities up to full scale when available should be used to evaluate code scale-up capability and accuracy to model important phenomena, whereas IET test data should be used to evaluate **overall** code accuracy.

10. **Determine Effect of Scale**: Differences for similar physical processes, but at different scales, should also be quantified for bias and deviation to establish a statement of potential scale up effects. In addition, separate biases should be evaluated as appropriate.
11. **Determine Effect of Reactor Input Parameters and State**: The effect upon uncertainty because of an imprecise knowledge of the reactor state and operating conditions at the initiation of the transient should be quantified. A typical example of these contributors is the potential variability of the fuel state that arises from the assumed condition of the fuel as a function of the burn-up cycle and the original manufacturing tolerances. Realistic variations are determined by examination of the most probable reactor state and the distribution around this condition using both experimental data and analytical studies. In addition, separate biases should be evaluated as appropriate.
12. **Perform NPP Sensitivity Calculations**: The influence of the individual contributors to uncertainty, determined in steps 9, 10, and 11, upon the primary safety criteria should be determined by performing NPP sensitivity calculations. That is, the individual variabilities cast in terms of bias and distribution are input to the NPP model and are used to determine their effect upon the uncertainty of simulating the primary safety parameters. These results are used in combining the biases and uncertainties for a singular statement regarding total uncertainty in steps 13 and 14.
13. **Combine Biases and Uncertainties**: Uncertainties determined in the above steps should be combined in a statement of total uncertainty. As there are several ways of combining them (addition, root mean square, response surface, etc.), a justification should be provided for selecting a particular method. Separate biases must also be added to produce total mean and 95% probability values of the appropriate parameter(s) including combined biases.
14. **Determine Total Uncertainty**: A statement of total uncertainty for the code may be given as an error band or statement of confidence about the code calculation with respect to the primary safety criteria (for example, PCT during a LBLOCA).

It is important to note that the CSAU methodology outlined above is not fully prescriptive regarding the details of its implementation. Rather, the CSAU methodology provides a complete and logical structure to which the details of alternative implementation approaches can and must be referenced and evaluated for completeness. In the following section, the approach selected by the NRC-organized Technical Program Group (TPG) to demonstrate the CSAU methodology is briefly described. During the process of developing the demonstration, the TPG explored many approaches; some proved fruitful and some did not. However, valuable lessons were learned in each case. The lessons learned are documented in the CSAU final report (Ref. 15) and should prove useful to those interested in alternative implementation approaches.

IV. CSAU DEMONSTRATION

An application of the CSAU methodology has been demonstrated by the TPG for a cold-leg LBLOCA in a Westinghouse four-loop PWR with 17 x 17 fuel. The demonstration conformed to the requirements of the revised 10CFR50.46 (Ref. 1). The BE PCT and uncertainty in predicting PCT were quantified. Only one of the acceptance criteria was evaluated, the PCT.

This focus was warranted as long as the total mean PCT including combined biases at the 95% probability level remained below the initiation temperatures for cladding oxidation and hydrogen generation from chemical reaction of the cladding with water or steam. Activities and results related to this demonstration are summarized below.

Element 1: Requirements and Code Capabilities (Steps 1–6, Fig. 2)

For the specified scenario (LBLOCA) and selected power plant, key phenomena and processes were first identified and then ranked by expert opinion and a subjective decision-making process, the Analytical Hierarchical Process (Ref. 16). The results were then summarized in a PIRT. The key phenomena and processes identified as important during the entire LBLOCA transient were break flow, stored energy and fuel response, reactor coolant pump two-phase flow, steam binding, ECCS bypass, and non-condensible gas (NCG). The steam binding and ECCS bypass phenomena are of importance only during the reflood phase of a LBLOCA.

The frozen code selected was TRAC-PF1/MOD1, Version 14.3. Exclusive of the many assessment reports, only one of which is cited here, the applicable code documents are Refs. 17–20. These documents were reviewed and the ability of the code to simulate these key processes was confirmed. As this effort was completed, available parameters and processes within the code related to the highly ranked phenomena were identified. For example, the parameters available within the code for modeling stored energy and fuel response were the gap conductance, peaking factor, fuel conductivity, surface heat transfer, initial power, clad conductivity, fuel and clad heat capacity, and pellet power distribution. Several code-related deficiencies were identified, e.g., the code does not provide models for dissolved NCG. The processes and outcomes associated with the first CSAU evaluation methodology element, requirements, and code capability are summarized in Fig. 3.

Element 2: Assessment and Ranging of Parameters (Steps 7–10, Fig. 2)

The processes and outcomes associated with the second CSAU evaluation methodology element, assessment, and ranging of parameters, are summarized in Fig. 4. A test data matrix was established to a) assess code accuracy to calculate the important processes shown in Fig. 3, and b) address code deficiencies identified in step 6.

The nodalization of the NPP was guided by past experience, by the need to capture the important phenomena and the design characteristics of the plant, and by the desire to perform NPP calculations in a timely and cost-effective manner. Subsequent assessment using SET and IET data were performed using the same nodalization.

Where possible, auxiliary calculations not requiring application of TRAC-PF1/MOD1 were used to further reduce the number of uncertainty parameters associated with the highly ranked phenomena identified in step 3, Fig. 2. For example, a closed-form fuel-rod model was developed at Brookhaven National Laboratory (BNL) and used to further reduce the set of stored energy and fuel response related parameters requiring evaluation to quantify individual contributions to uncertainty (Ref. 21); the reduced set of parameters was the gap conductance, peaking factor, fuel conductivity, surface heat transfer coefficient, and the minimum homogeneous nucleation temperature (T_{min}). This process is illustrated in Fig. 4. For these remaining parameters, uncertainty ranges were determined and used later in the NPP calculations (step 12, Fig. 2). Where such simplified models could not be identified, code

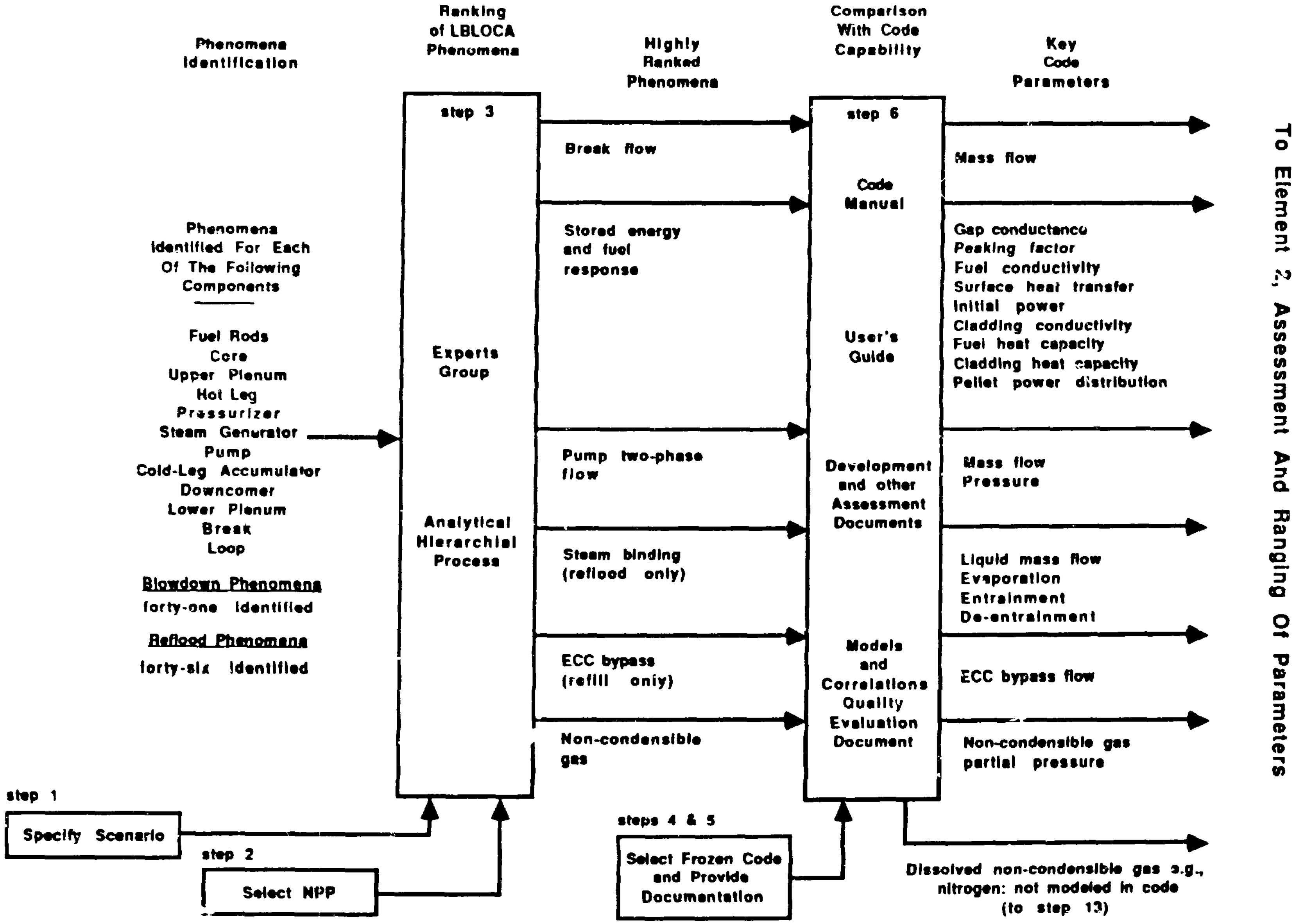

Fig. 3.

Illustration of Element 1, Requirements and Code Capabilities, of the CSAU Methodology as applied to a LBLOCA and TRAC-PF1/MOD1.

assessments were performed using the SET and IET data. Such assessments proved useful in two ways, one direct and one indirect.

First, data were used directly to develop uncertainty ranges for the selected code parameters. The individual uncertainty contributions of previously identified parameters relative to selected data were determined (steps 9 and 10, Fig. 2) and later input to the NPP model so that the effect upon the primary safety criteria (PCT in the demonstration) could be evaluated (step 12, Fig. 2). For example, the scale-related bias in the TRAC-PF1/MOD1 prediction of ECC bypass as measured by the lower-plenum filling rate was examined at BNL (Ref. 22). BNL examined the data from Creare 1/15 and 1/5 scale downcomer tests, the Battelle Columbus Laboratory 2/15 downcomer tests, and UPTF 1/1 scale downcomer tests. Based on these studies, it was demonstrated that a scale effect exists in the code when modeling the ECC bypass. TRAC was shown to overpredict the delivery of ECC to the lower plenum at smaller scales and underpredict the delivery of liquid at full scale. The BNL work provided convincing evidence of a scale-related bias in TRAC regarding the prediction of ECC bypass phenomena. This code bias was subsequently quantified using a bounding argument and the result factored into the overall quantification of uncertainty as an additional margin (step 13).

Second, the data were used in a supportive role to provide insight into the accuracy of code calculations and to confirm conclusions that were being drawn regarding specific CSAU studies. LBLOCA transients have been run in reduced-scale integral and separate effect facilities such as Semiscale, the Loss-of-Fluid Test Facility, the Cylindrical Core Test Facility, and the Slab Core Test Facility. Numerous assessment calculations have been performed using the data from these tests. From such calculations it is possible to state that a given code, e.g., TRAC-PF1/MOD1, will predict a selected parameter in the sub-scale facility, such as PCT or cladding quench time, with a stated bias and within a stated uncertainty band. In addition, the level of confidence related to such calculations can be provided. Although such results provide insight into the ability of the code to calculate similar phenomena in operating plants, they do not, in themselves, transfer directly to the full-size plant. Therefore, such results are considered to be supportive within the context of the CSAU methodology. It is important to emphasize, however, that the availability of such supportive results is important. In addition to increasing confidence that the dominant phenomena are modeled in the code, the quantified uncertainty for full-size plants can be checked for both trends and magnitudes as insurance that problems in application of the CSAU method at full scale do not go unrecognized.

Element 3: Sensitivity and Uncertainty Analysis (Steps 11–14, Fig. 2)

The approach taken to quantifying the total mean and total 95% probability PCTs, including margins, is illustrated in Fig. 5. Inputs for NPP calculations come from several sources. One source of uncertainty is related to code, scale, and experimental data contributors to uncertainty (e.g., pump and break flow characterization, core heat transfer coefficient calculation, and T_{min} calculation), which are evaluated as part of the second CSAU evaluation methodology element (steps 9–10, Fig. 2); the ranges and distributions of these parameters were used to specify one set of NPP calculations. Another source of uncertainty is the range and distribution of plant operating conditions and process variations that arise from imprecise knowledge about the reactor state during the transient. For example, the ranges and distributions of the fuel conductivity, gap conductance, and peaking factor as a function of the

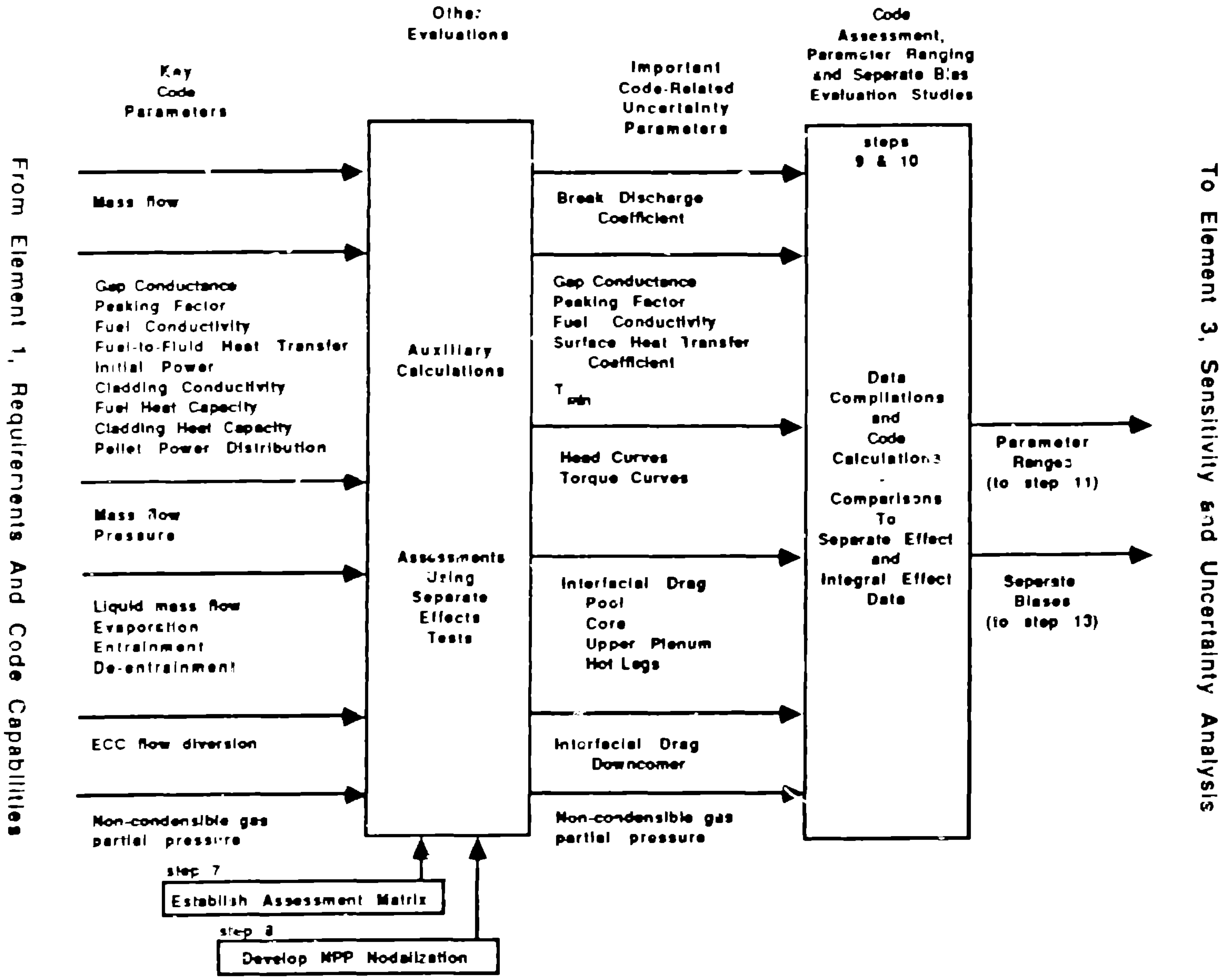

Fig. 4.
Illustration of Element 2, Assessment and Ranging of Parameters, of the CSAU Methodology as applied to a LBLOCA and TRAC-PF1/MOD1.

burn-up cycle and the original manufacturing tolerances were determined and used to specify a second set of NPP calculations (step 11, Fig 2). The result of each NPP run was a PCT related to the particular parameters specified.

To combine these individual uncertainties (step 12, Fig. 2), a probability distribution function (pdf) was generated from Monte Carlo sampling of a response surface representing the code output from the NPP calculations described above. The response surface methodology requires that a pdf be specified for each of the significant parameters; a uniform distribution was chosen because investigation of the literature does not indicate that the uncertainty of any of the parameters exhibits a particular distribution in its experimental data. Also, a uniform distribution requires the least prior knowledge (Ref. 23). For the CSAU demonstration, 184 calculated PCT total points arising from TRAC-PF1/MOD1 calculations were used to construct response surfaces. Many of the points arose from cross-product calculations: 22 of the 184 points were single parameter variations, 98 were double, 57 triple, and 7 quadruple. For the blowdown peak, response surfaces were constructed for seven significant LOCA parameters: peaking factor, gap conductance, fuel conductivity, fuel-to-fluid heat transfer coefficient, break discharge coefficient, pump characteristic, and T_{min}. For each response surface, a process of random sampling or Monte Carlo was used to get the values of mean PCT, the standard deviation, and the PCT at the 95th percentile probability level. A 50,000 history Monte Carlo sample appears to provide an acceptable degree of convergence. The statement of combined uncertainty by pdf includes the specification of a mean PCT and the PCT at 95% probability. For the CSAU demonstration, it has been concluded that a TRAC-PF1/MOD1 prediction of the PCT during a cold-leg LBLOCA in a Westinghouse four-loop PWR with 17 x 17 fuel will be equal to or less than 1447°F (1059 K) 95% of the time. This PCT occurs during blowdown.

However, this is not a final statement of the total uncertainty (mean and 95% probability PCT) until appropriate separate biases are incorporated. As the effects of some contributors to uncertainty were not quantified by means of sensitivity calculations and response surface (because of limited data base, considerations of cost effectiveness and/or scheduling, etc.), separate biases were evaluated based on bounding calculations. These separate biases are included in the total uncertainty as shown in Fig. 5. For the CSAU demonstration, five separate biases were evaluated in this manner.

First, a bias was quantified to account for hot channel effects that could be modeled with the code but were not modeled for economic reasons. Second, a bias was estimated for uncertainty related to modeling phenomena associated with the presence of NCG. The TRAC-PF1/MOD1 code includes the capability for modeling NCG as a separate fluid component, e.g., following the movement of the accumulator cover gas before and following accumulator injection. However, the code does not include models for dissolved NCG. Because the code lacked this modeling capability, the bias associated with dissolved NCG was analyzed and bounded. Third, a bias was taken for a code error, a nonconservative implementation of a heat transfer correlation that did not adequately represent scale effects. Fourth, a bias was taken by bounding ECC bypass effects to account for code correlations that did not adequately represent scale effects. Fifth, a bias was taken for steam binding effects because the code did not adequately model core entrainment processes.

The total uncertainty in PCT is obtained by adding the combined uncertainty by pdf and the additional separate biases described above. For the CSAU demonstration, the contributors

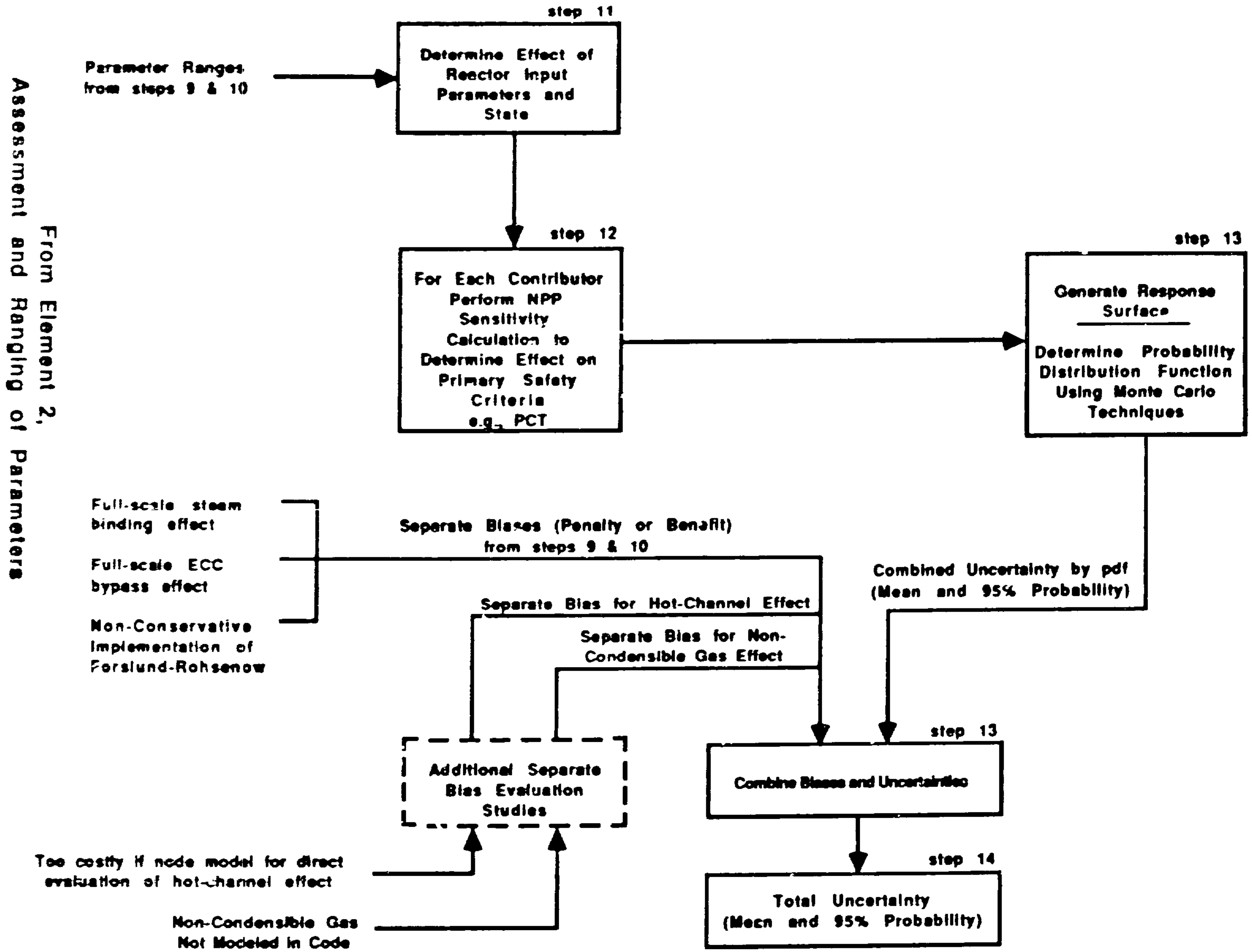

Fig. 5.
Illustration of Element 3. Sensitivity and Uncertainty Analysis, of the CSAU Methodology as Applied to a LBLOCA and TRAC-PF1/MOD1.

to the total uncertainty are presented in Table I. From Table I it can be seen that the mean and 95% probability PCTs, excluding separate biases, reach maximums during the blowdown, and lesser peaks are predicted for the first and second reflood peaks. After including the combined biases, the total mean PCT still occurs during reflood. However, the total 95% probability PCT including the combined margins occurs during the second reflood peak and is 1572°F (1129 K). This shift in the peak to the latter part of the LBLOCA transient is a reflection of the increasing uncertainty in PCT as time increases and the increasing importance of several separate biases later in the transient. In this regard, the two key contributors are the nonconservative implementation of the Forsland-Rohsenow correlation and steam binding effects. Each of these margins is related to code model and correlation defects and could be eliminated by improving the appropriate code models and correlations.

SUMMARY

The commissioners of the NRC recently approved the final version of a revised rule for the acceptance of ECCS. The revised rule permits alternate ECCS performance analysis, based on BE methods, to be used provided the licensee quantifies the uncertainty of the estimates and includes that uncertainty when comparing the calculated results with prescribed acceptance limits. Under NRC sponsorship, a CSAU evaluation methodology has been defined and a demonstration completed for a Westinghouse four-loop PWR with 17 x 17 fuel. The demonstration considered a cold-leg LBLOCA and calculated the total uncertainty associated with use of the thermal hydraulic systems code, TRAC-PF1/MOD1, Version 14.3.

The demonstration effort showed that uncertainties in the complex phenomena occurring during accident conditions in NPPs can be quantified. The demonstrated methodology is auditable, traceable, and practical in the sense that sound engineering judgment, accompanied by external peer review, is an integral part of the process. The demonstration results confirm the existence of a significant margin in current plant operating conditions for the demonstration NPP.

ACKNOWLEDGMENTS

Development and demonstration of the CSAU evaluation methodology was a team effort by the TPG-member authors of this paper and their co-workers at the participating institutions. The lead author of this paper wishes to acknowledge that much of the information presented in this paper was extracted from CSAU documentation prepared by other members of the TPG. In particular, the significant contributions by Idaho National Engineering Laboratory staff members G. E. Wilson and K. R. Katsma are gratefully acknowledged.

TABLE I

TOTAL UNCERTAINTY FOR THE CSAU DEMONSTRATION*

	PCT (F)**		
	BLOWDOWN	REFLOOD	
		1st Peak	2nd Peak
Mean PCT (combined uncertainty by pdf)	1162(901)	978(799)	758(676)
95% probability PCT (combined uncertainty by pdf)	1447(1059)	1399(1032)	1336(997)
Mean separate biases added for:			
Hot-channel effects	63(35)	25(14)	-14(-8)
NCG effect (dissolved nitrogen)	N/A	18(10)	18(10)
Nonconservative implementation of Forsland-Rohsenow correlation in code	47(26)	84(47)	160(89)
Full-scale steam binding effects	N/A	-9(-5)	106(59)
Full-scale ECC bypass effects	N/A	-34(-19)	-34(-19)
Combined mean biases	110(61)	84(47)	236(131)
Total mean PCT (including combined biases)	1272(962)	1062(845)	994(807)
Total 95% probability PCT (including combined biases)	1557(1120)	1483(1079)	1572(1129)

*Application to a four loop Westinghouse PWR for a cold-leg LBLOCA and using TRAC-PF1/MOD1, Version 14.3.

**Numbers shown in parenthesis are PCT in K.

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