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STATUS OF CODE UNCERTAINTY EVALUATION METHODOLOGIES

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ABSTRACT

During recent years there has been increasing demand for quantification of the uncertainties associated with realistic predictions of the thermal hydraulic behaviour of nuclear power plants during various accident and transient conditions. Uncertainty quantification is especially useful in allowing use of realistic calculations with "best-estimate" (BE) codes (instead of conservative calculations with "evaluation model" (EM) codes) in licensing analyses. BE codes combined with their application uncertainties provide the technical basis for reductions in calculated safety margins where considered excessive.

This paper is based on an paper presented at a CSNI Specialist Meeting in April 1992, which described and compared three major code uncertainty evaluation methodologies: one (CSAU) by USNRC, USA, another by AEA Technology, UK, and the third by GRS, Germany. It focuses on the progress that has taken place during 1992 and 1993, and also includes very short descriptions of the recent related work in France and Italy. All of the methodologies are still under development, except for the essentially complete CSAU Method. The first major developmental applications of the Winfrith and GRS methods, and several new applications of CSAU have been completed, and are summarised here.

1 FOREWORD

The assessment of "best-estimate" (BE) codes aims at providing sufficient evidence that they can be used reliably and accurately enough, within their intended ranges of application, for realistic predictions of the thermal-hydraulic behaviour of nuclear power plants (NPPs) during various accident and transient conditions. So far most assessment has been qualitative, with few exceptions, but during recent years there has been increasing **demand for quantification** of the associated uncertainties, especially in conjunction with efforts to allow use of realistic calculations with BE codes (instead of conservative calculations with "evaluation model" (EM) codes) in licensing analyses, and to allow reductions in calculated safety margins where considered excessive. However, quantification of the uncertainties is generally desirable, and often necessary, in many other applications as well.

The first efforts to develop methodologies for quantitative uncertainty evaluation were started in the USA in the 1980's, and resulted in the USNRC Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology. In the latter half of the 1980's development of two other methods began at GRS, Germany, and at Winfrith, UK. Their principles were quite similar, but differed rather clearly from those of the CSAU. Considerable progress has been made recently, and interesting experiences reported. In addition, related, mostly complementary work has also been started at the French research centres in Fontenay-aux-Roses, Grenoble and Cadarache, and at Pisa University, Italy.

This paper is **based on the paper "Quantification of Code Uncertainties"** presented at the CSNI Specialist Meeting on Transient 2-Phase Flow in April 1992, which described and compared the CSAU, GRS and AEA methods as they stood at the end of 1991. **Figure 1** shows a comparison of their main steps. The focus is now on the progress that has taken place during 1992 and 1993. Very short descriptions of the recent developments in Italy and France are also included. It may be noted that this paper can only be considered as a short summary due to the 10 pages limit.

The methodologies aim at addressing all **sources of uncertainty** in code predictions, but for practical reasons all of them are not being considered in the central quantification phase. Those uncertainties that are impossible or difficult to take into account through parametric variations are addressed early in the process when determining the applicability of the code. At this stage it will be checked e.g. whether the code is generally stable and reliable without non-physical discontinuities etc., and has the necessary models to address all relevant phenomena, and produces qualitatively good results according to the assessment carried out. Also, some of the uncertainties are neglected or considered by combination

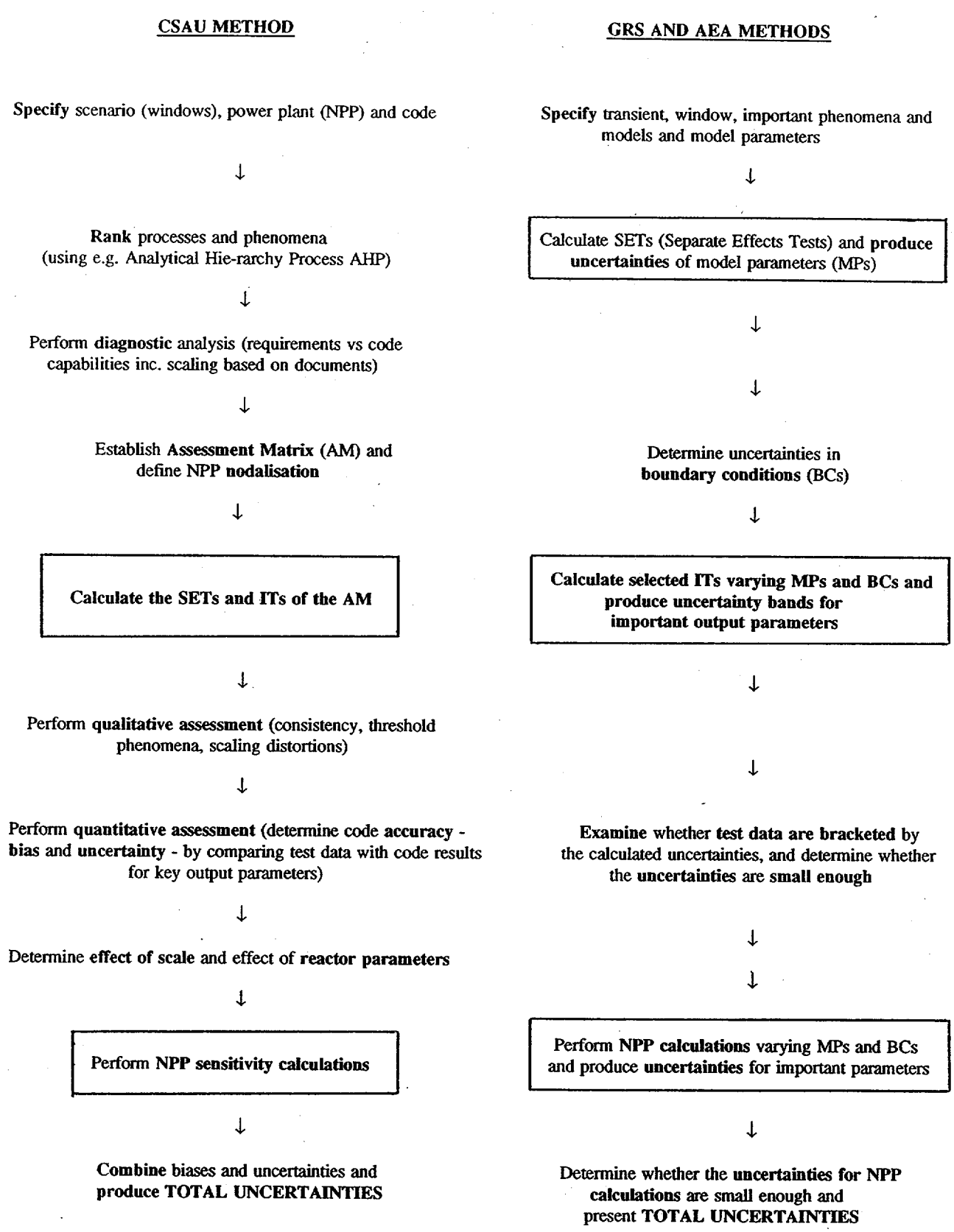

Figure 1 Comparison of Three Code Assessment Methods

as a result of a ranking of the phenomena. The number of uncertain parameters addressing the important phenomena being finally considered, however, is strongly dependent on the method.

Quantitative **uncertainty evaluation** is not the same as code **assessment** or **validation**, although a comprehensive uncertainty evaluation methodology may include the latter as one phase, which should be performed before applying the method to a particular case like (most importantly) a nuclear power plant. Essential in uncertainty evaluation are sensitivity calculations where input/model parameters are varied in order to study the effect of their variability on the uncertainties of the most important results. Uncertainty evaluation is also different from code **accuracy evaluation**. Accuracy evaluation methods aim at quantifying the agreement of code calculation results with the measured quantities (during an experiment). An accuracy evaluation method may be developed into a **quantitative assessment** method by introducing a way to combine the accuracies of several important quantities, and the results of many assessment calculations. It may also be developed further into an **uncertainty evaluation** method by adding a method to extrapolate the quantitative results of the assessment to NPP predictions.

2 RECENT APPLICATIONS OF THE USNRC CSAU METHODOLOGY

Recent applications of the CSAU (Code Scaling, Applicability and Uncertainty) Methodology have provided insights regarding **questions** like: 1) Is the CSAU sufficiently general to be applicable to other codes, scenarios and NPP designs (CSAU is basically a prescription for a process with wide flexibility)? 2) Is it economically feasible to apply CSAU to a larger number of scenarios of a Nuclear Power Plant (NPP) operating envelope? 3) Is the expert opinion used in the CSAU sufficiently accurate to justify making licensing/regulation decisions based on the uncertainties determined? and 4)Is further validation of the numerical uncertainty values necessary?

One of the recent CSAU applications aimed at adapting and demonstrating the CSAU methodology for a **Small Break Loss-of-Coolant-Accident (SBLOCA) in a Babcock and Wilcox (B&W) design** using RELAP5/MOD3 as the simulation tool. It was assumed e.g. that one of the high pressure injection (HPI) lines would break near its connection to the cold leg, and the operator would act according to emergency operating procedures (defined as part of the scenario, not as a sensitivity code parameter).

The **minimum vessel liquid inventory** was selected as the main Primary Safety Criterion (PSC)(key output parameter). Plausible phenomena were identified and then ranked for their influence on the PSC. The procedure used **two independent panels** of experts. Both panels divided the transient into 4 phases (blowdown, natural circulation, loss of natural circulation and boiler-condenser mode). For each phase the NPP components were evaluated according to their influence using the Analytic Hierarchy Process (AHP), and for each component and for each phase, the phenomena pertinent to each component were ranked. The significant differences between the results of the two panels were resolved on the basis of experimental and/or analytical arguments, and a composite Phenomena Identification and Ranking Table (PIRT) was produced. The following **phenomena** (and their associated **components**) were selected as important enough to be considered in the NPP sensitivity calculations: 1) Break Flow (Break), 2) Natural Circulation (Vessel & Steam Generator), 3) Decay Heat (Core), 4) RCP Performance (RCP), 5) ECCS Flow (HPI), 6) Steam Generator Heat Transfer (Steam Generator), 7) Phase Separation in the Candy Cane (Candy Cane), and 8) Reactor Vessel Vent Valve Performance (Vessel). Eight sensitivity parameters (SPs) were then selected to represent these phenomena.

Thirty three **sensitivity calculations** varying the SPs over their relevant ranges (based on a data base consisting of 14 SETs and 7 IETs) were used to build a response surface. The results indicate that for the parameter uncertainties used core uncovery with fuel cladding heatup is highly unlikely. The result was very robust with respect to the uncertainties of the SPs, but would apparently be much more sensitive to operator actions.

During the **discussions** within the CSNI Principal Working Group 2 (PWG2) several critical comments were made with regard to this study: A more challenging SBLOCA case should have been chosen. Almost nothing of interest happened during the transient (no core uncovery). Only one single-valued output parameter was chosen for the uncertainty evaluation. No continuous-valued parameter was chosen in this or any other case, indicating the inability to determine the uncertainties for such parameters. Restriction to single-valued parameters, and only one or two of them, may oversimplify the problems and lead to wrong conclusions. The small number of calculations reported to be adequate has been questioned as well. As a specific detail in this study, the chosen range for the break discharge coefficient (C_d) of 1.1-1.5 seems unrealistic. The range was based on only two Marviken tests. Usually C_d varies around 1.0.

The objective of another application was to develop **PIRTs applicable to a Large Break LOCA in a heavy water production reactor** (Savannah River Site). They provide the basis for improvement in the thermal-hydraulic analytic methods used to perform safety analyses, including need for new experimental data, and ultimately were used to focus the validation of these codes. The **first three steps of CSAU** (specify scenario, select NPP, and identify and rank processes) were applied to identify the plausible phenomena and processes, and prioritize them with respect to their importance to the system response, in the context of the safety criteria. PIRTs were established at the third step.

In this case **three expert panels** were used, working independently, to formulate three sets of PIRTs. Based on experimental and analytical evidence and the objectives of the work, the onset of significant voiding was judged to be the most appropriate safety criterion for the first phase, and the onset of dry-out for the second phase. Subsequent to the formulation of the three independent sets of PIRTs, the results were compared, and significant differences were resolved primarily on the basis of technical arguments and reconciliation of differences in the approach. In the few cases where consensus was not reached a conservative approach was taken. Phenomena which were considered to be plausible by one or more panels were retained. Phenomena ranks (1 to

9) which differed by one or more units were assigned their highest rank. 26 phenomena (10 components) from the PIRTS were selected as the dominant ones.

The objective of the third application case was to determine the **dominant thermal-hydraulic phenomena for a Modular High Temperature Gas-Cooled Reactor (MHTGR)** during representative normal, off-normal and accident conditions. The results provide the basis for evaluating the applicability of existing codes, define the requirements for the development of new ones, and also support the quality assurance related verification and validation of the codes. The **first three steps of the CSAU** methodology were again used. The requirements were formulated in terms of their relative importance to satisfy the operational, investment and/or safety criteria. A minimum set was defined to capture (almost) all of the plant operating space, including credible accident conditions (probability $>10^{-8}$).

Thirteen basic scenarios were found to contain the necessary elements to well represent the credible events. That is, validation of a code with the 13 basic scenarios would cover all credible events. 34 of the other credible scenarios were judged to be phenomenological identical to the basic scenarios, and 40 to be similar to one or more of them. Certain events composed of highly improbable combinations of postulated multiple failures may require additional consideration, but it may be more cost effective to use bounding benchmark calculations, etc., rather than validation of the code(s) for these events.

The objective of the fourth application case was the evaluation of the **importance of components and phenomena in the thermal-hydraulic safety analysis of the proposed experimental Advanced Neutron Source Reactor (ANSR)**. In this context, in the absence of sufficient experimental data, the probable uncertainty of the ranking of components and phenomena was evaluated by performing a **statistically based uncertainty analysis**. The uncertainty was estimated by assigning probability distribution functions (pdf's) to the potential errors in the initial expert estimates of pair-wise importance between the attributes, and propagating them through the Analytical Hierarcy Process AHP (using a Monte Carlo sampling technique) to determine pdfs of uncertainty in the ordered ranks of the attributes, and of their importance levels (low, moderate or high). Additional tests were performed to determine the sensitivity of the results to the assumed error pdf, Monte Carlo sample size, the AHP solution scheme, and numerical round-off error in the computing system.

It was **concluded** e.g. that the AHP is a robust methodology to determine nuclear reactor component (and phenomena) importance in operational, investment and safety studies, for fairly small matrices (down to six elements), and that with regard to application to the ANSR, there was zero probability of importance level changes of two (high to low or low to high). On the average, the probability for no change in importance level was greater than 87%.

With regard to the important **CSAU applications currently in progress**, the USNRC and its contractors are using the CSAU methodology to **focus experimental and analytical research to support certification of the new advanced light water reactor (ALWR) designs**. For the earlier exercises described above, **six conditions** needed to be satisfied before applying CSAU: 1) The computer code was "frozen", 2) The code documentation was complete, 3) A single transient or closely related set of transients were the subject of assessment, 4) The safety evaluation criteria were established, 5) The experimental assessment matrix was complete, and 6) Sufficient information and a cadre of experts existed to construct a valid PIRT. These conditions could not be met for the "passive" ALWRs, and a **modification (figure 2) to the process** was devised, which accommodates the changing nature of the codes, development of an experimental data base, and development of additional information and expertise. The numbering system in the figure is based on the original CSAU (OCSAU) and is, therefore, not sequential.

The OCSAU process is linear in structure except for iteration of experimental assessment analysis to determine the appropriate NPP noding. The proposed ALWR CSAU process is very iterative prior to Step 9 to satisfy the conditions necessary to quantify the uncertainties. For the ALWR process, the following **6 questions are posed at Step 6** to determine if further experiments, development assessment, and plant analysis are needed (In case of one or more negative answers Step 2 has to be revisited):

1. Are the input decks consistent and appropriate? In the OCSAU, this question was limited to the noding adequacy.
2. Is the code applicable to the analysis of all desirable scenarios? This alone was Step 6 in the OCSAU. If the code is not applicable, adequate, or the assessment results are unsatisfactory, corrective action is required.
3. Should new scenarios for assessment be considered? Experience has confirmed the need for this question.
4. Have sensitivity studies, experiments and assessment confirmed the validity of the PIRT? If not (new experimental data?), re-ranking should be done.
5. Are additional experiments needed? The PIRT process and current capability of the code may indicate a need for additional SETs or IETs.
6. Is the documentation adequate for peer review/uncertainty determination?

During the **iteration** process, steps may be worked in parallel or even omitted, but on the last iteration, prior to uncertainty quantification, all steps should be satisfactorily completed and the above questions resolved. Then the result should be equivalent to the OCSAU, and the process common from Step 9 onwards. It may be noted that the methodology summarised in figure 2.1 is still undergoing development and change. It will be finalized and documented in 1994.

Based on the results of the above CSAU applications, and in the context of the three questions posed at the beginning of this section it has been **concluded** that: 1) As a process, CSAU has been demonstrated, although not conclusively, to be sufficiently general to be applicable to other codes, scenarios and NPPs. 2) The more accurate the uncertainty determination the more costly the analysis; generally increased accuracy and reduced level of uncertainty are directly correlated. The costs of using CSAU have decreased as experience has been gained with its application.

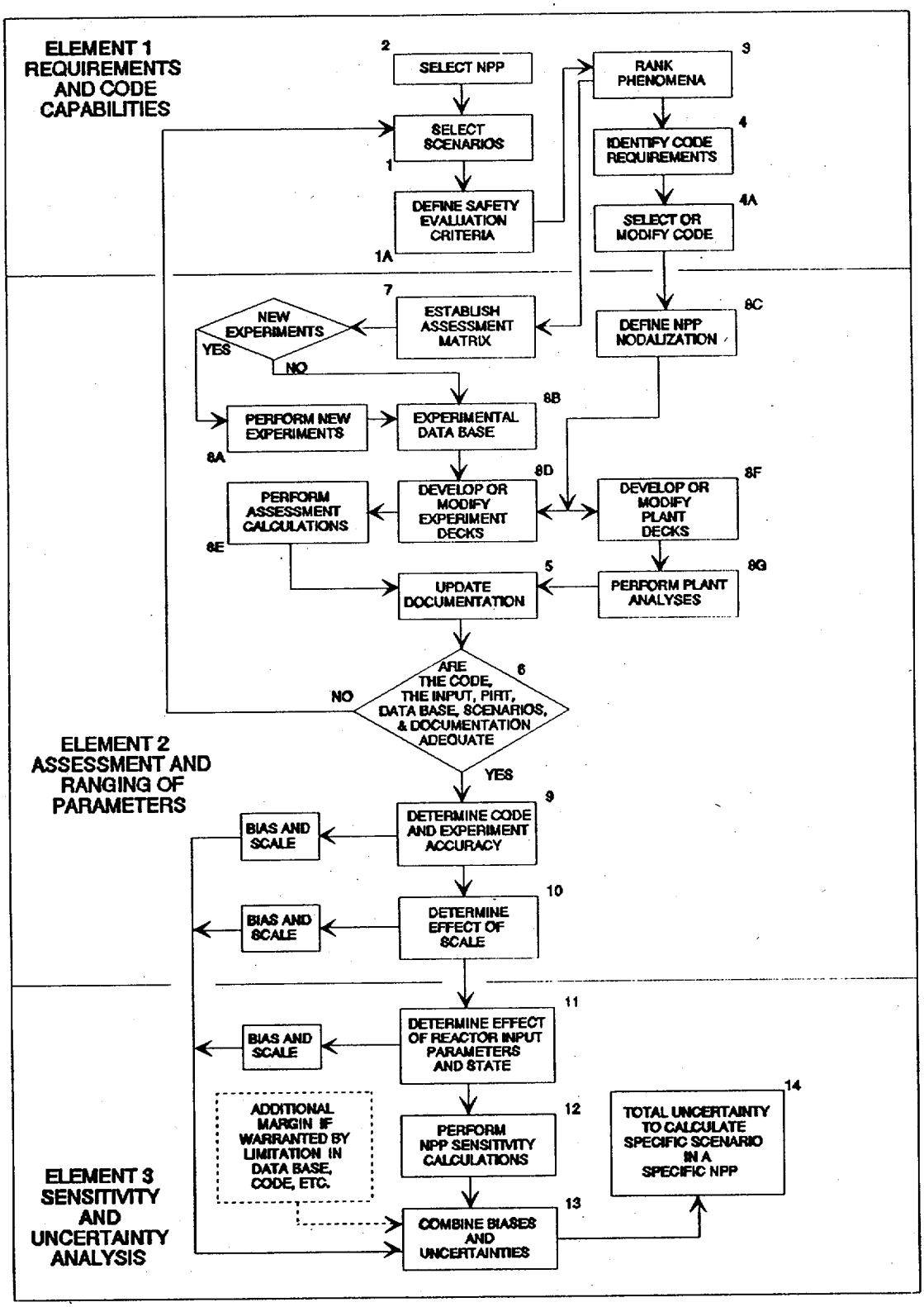

Figure 2 CSAU evaluation methodology applied to ALWR development, assessment and analysis.

This is due to increased experience and the advances in workstation analysis. 3) The results of the study to statistically evaluate the uncertainty embedded in the use of expert opinion and the AHP appears to confirm their sufficient accuracy. With regard to the larger question of validating the accuracy of the final uncertainty, experimental evidence should be shown supporting the data used throughout any application process. Increased knowledge of the accuracy of CSAU could be gained by a comparison with the results of an identical application of the other methodologies either by applying the various methodologies to an experiment in a test facility and/or to a postulated scenario in a full scale NPP.

In addition to the comments presented above in conjunction with the B&W SBLOCA study, the following kind of **generic comments** have been made during **discussions** within the CSNI PWG2: Although the effort required to perform a CSAU application has been reduced as a result of experience gained and of the progress in computer technology, the effort is still being regarded as large, and a further reduction would be very desirable. Application of (parts of) the CSAU to specific purposes like revealing weaknesses and shortages in both the codes and the available assessment data bases to facilitate planning of future research strategies, and modifying the block diagrams accordingly has been generally welcomed. In fact, adding feedback loops in the original diagram is recommended even if their use is not currently expected in the case of application to current LWRs. AHP and PIRT are considered generally valuable, but require a large effort and retain a subjective nature. Addition of biases seems to be inconsistent with the basic method. The number of necessary calculations is not given.

Although the CSAU applications have demonstrated the generality and usefulness of the basic method, the accuracies of the final uncertainties have not been fully validated. Concerning this question, the authors of this paper recommend the validation and comparison of all the methods as an international activity (see Chapter 6 "Conclusions").

3 RECENT APPLICATIONS OF THE GRS METHODOLOGY

A program package has been developed to support the GRS uncertainty and sensitivity analysis method. The techniques are primarily based on statistical tools to **specify uncertain input parameter variations and to evaluate the uncertainty ranges and distributions of selected code results**, and also to determine the ranking of the influence of individual input uncertainties on these results. These are available simultaneously for all single- and continuous-valued (time-dependent) output quantities of interest. **Figure 3** shows a flowchart of the central part of the method.

A major task is to specify the **input uncertainties** (for initial/boundary conditions, model parameters, model equations, and nodalisations), and to represent **modeling uncertainties** by additional (model) parameters. Both their range of values and their probability distributions within that range must be defined dependent on the state of knowledge. Also dependencies between the parameters have to be identified. As much as possible, these specifications have to be based on experimental data, code validation and other experience. The knowledge of well experienced code assessors is essential for this part of the analysis.

The computational effort for the GRS method is independent of the number of uncertain parameters, but dependent on both the desired probability fractile and confidence level, i.e. on the statistical tolerance limits. For example, a minimum of 59 computer runs is necessary for a 95 % subjective probability and 95 % confidence limit based on the identified and quantified uncertainties (i.e. to be at least 95 % confident that at least 95 % of the combined influence of all quantified uncertainties and their state-of-knowledge quantifications are below the tolerance limit).

So far the GRS method has been **applied** to several nuclear and non-nuclear **safety related computer code calculations**, e.g. to the following **thermal-hydraulic** cases: 1) Investigations of computed fog formation rate and aerosol behaviour in a containment test facility, 2) Applications of a condensation model, and 3)A separate effects test (SET) calculation with the ATHLET code.

The SET experiment was the French **OMEGA Rod Bundle Test N°9, a blowdown with a PWR-type rod bundle** consisting of 36 electrically heated rods of 3.66 m heated length connected to spherical plena (D_i of 0.5 m) with convergent horizontal O.198 m long break nozzles. The initial pressure was 13 MPa. The five equation version of ATHLET was used. It contains two mass conservation equations and two energy equations (for steam and water) and one momentum equation (for the mixture) in combination with a drift-flux model.

After identifying the relevant phenomena a total number of **60 uncertain input parameters** was selected for the analysis. These consist of 40 model parameters, 11 parameters for selection of different correlations, 3 properties of heater rod (Inconel), 2 parameters for closing times of hot and cold side valves, 1 parameter for heater power, 1 parameter for heater power shut-down, 1 numerical parameter, 1 parameter for noding change.

The following **31 key output parameters** were selected for determination of their uncertainty ranges and sensitivity measures: PCT, time of PCT, total CPU time (single values), 7 clad temperatures, 6 fluid temperatures, 6 pressures, 6 mass flow rates, and 3 void fractions (time dependent values).

100 ATHLET code calculations were performed with randomly selected values of uncertain input parameters within their specified ranges and according to their probability distributions. The total CPU time was 20 h on a CONVEX computer. The problem time for each run was 40 s. Some (14) model parameters (uncertain correction factors serving as means for treating model uncertainties) had to be made available via the input deck.

The **uncertainty range** of the calculated PCT was from 788°C to 1078°C with 95%/95% statistical tolerance limits. The highest measured PCT was 914°C taking into account the thermal inertia of the thermocouples (diameter of 1 mm compared with cladding thickness of 0.5 mm in the hottest region). The first direct measurement had been 780°C. **914°C is equal to the estimate of the 56 % fractile of the resulting probability distribution of the PCT** (914°C will not be exceeded with a 56 % probability). The measured data for all other selected output values were within their calculated uncertainty ranges as well. The following input pa-

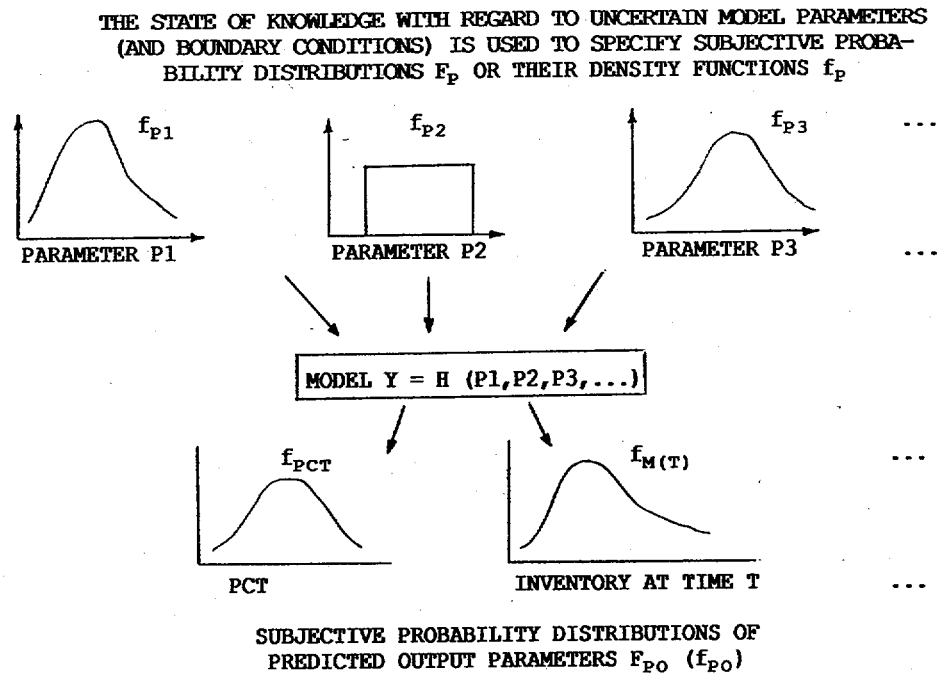

Figure 3 Flowchart of the Central Part of GRS Method

rameters were found to have the **largest contributions** to the PCT uncertainty: 1. Time of heater power shut-down (varied within 0.5 s), 2. Evaporation model, 3. Selection of drift-flux-model in vertical pipes, 4. Heater power, 5. Pressure loss coefficients in the bundle (spacers), 6. Minimum film boiling temperature correlation and 7) Specific heat capacity of the Inconel heater rod.

In **summary**, a large number of input parameters were included, the calculated uncertainty ranges were plausible, the rankings of individual input parameter uncertainties were determined, and a significant influence of experimental uncertainties were identified (heater power shutdown, value of heater power, heat capacity of heater rod). The uncertainty analysis for this SET required an effort of 1.5 person-years. One of the most important lessons learned during this study was that the specification of input uncertainties is a very important step. Some of the input uncertainties initially judged to be of minor importance turned out be important after all. A very good knowledge of the validation and the models of the specific computer code is essential. More experience through analyses of SETs and IETs will be beneficial before applying the method to large NPPs.

Based on **discussions** within CSNI PWG2 it seems that the GRS Method relies the least on experts' judgment (no ranking of input parameters to reduce their number, random selection of their values within their ranges, and no use of of biases on key output results), the number of necessary calculations is independent of the number of both input and output parameters, it is applicable to both single- and continuous-valued parameters, and it provides individual sensitivity measures of input parameters on the output parameters. However, the number of calculations needed is rather high. The basis of the necessary number is determined by the required statistical tolerance limits, but has been questioned, as well as generally the application of statistics to non-random quantities originating from a non-ideal experimental data base and to parameters with no possibility of direct comparisons with measured data. The effort involved in determining the probability distributions of the input parameters is also considerable, but will probably be reduced drastically after the first applications. As with all the methods, the adequacy of the procedure to take into account all the sources of uncertainty satisfactorily is being questioned as well.

4 RECENT APPLICATIONS OF THE AEA METHODOLOGY

Two studies have been performed since 1986 to demonstrate the AEA methodology for TRAC-PF1 calculations of PWR LOCAs. The first one focused on a LOBI test (BL-02) and the second one on a PWR large break LOCA (LBLOCA).

The method aims at producing **reasonable uncertainty ranges** (RURs) for the output quantities of interest. A RUR has been defined as the smallest possible range of values that excludes all values for which there is reasonable certainty that they are inconsistent with all available evidence. Mathematical proof of the inconsistency is not considered possible. This means that **reasoning**, sufficient to satisfy peer reviewers knowledgeable in the area, must be given to justify the inconsistency.

Many of the quantities in LOCA analysis are "data-free" i.e. they have not been measured, and classical statistical methods are not considered applicable for them. Other physical arguments have to be put forward to convey reasonable

certainty. Use of the analysts' personal subjective probability distributions to describe uncertainty ranges, even when the quantities are not random quantities, is not considered appropriate either.

In selecting the important uncertainties to be examined the analyst must exercise judgement and use all available relevant information and perhaps preliminary calculations. The ones omitted may be rereviewed. Because most uncertainties are "data-free", the RUR for a quantity of interest (e.g. the PCT) is obtained by **maximising and minimising the quantity over the uncertainty space.** Because calculations are expensive this space has to be sampled sparsely.

The PCT will vary in a non-monotonic, non-linear and discontinuous way in various parts of the uncertainty space, and cannot be approximated by a polynomial response surface over the whole space. Pdf's for the uncertain quantities will not be available either. The most cost-effective method is considered to be obtaining the best possible understanding of the problem and the interactions of the processes, and using scientific and mathematical judgement iteratively to seek the overall maximum(s) and minimum(s).

Experience has shown that it is possible to tune large codes so that they can predict the results of experiments in the development data base more accurately than other data. This may be due to compensating errors. After the code and the method are refined so that uncertainty ranges for developmental data are narrow enough the code is applied to an independent data base. If these data are encompassed by the predicted uncertainty ranges the method is considered to be validated. Otherwise further refinement work is needed (**figure 4**).

In order to demonstrate the processes of uncertainty identification, quantification and combination before moving to a full plant calculation a **pilot study of the LOBI SBLOCA experiment BL-02**, was made. The uncertainty study was conducted "blind", and the code used was TRAC-PF1/MOD1 version 13.0 (with generalised heat structures modified to allow models to be varied from input).

Four input uncertainties were identified as important, and the uncertainties of **four calculated quantities** were determined: primary system coolant mass and coolant mass in the core (as functions of time), minimum coolant mass in the core, and the time of that minimum.

Ranges for the input uncertainties were derived from data. A base case and **13 calculations** in which one uncertainty was varied from the base case were run first. Then a **further six calculations** in which the uncertainties were varied together were run. When the results were compared with the measured data, the two functions of time were mostly bounded and the two discrete quantities were bounded by the calculated uncertainty ranges. The failures to bound data were attributed to omission of offtake modelling and possible understating of choked flow modelling uncertainty. The overall conclusion was that the method meets the requirements for an uncertainty study.

The **LBLOCA Study** investigated the uncertainty in PCT in a LBLOCA calculation for a 4-loop Westinghouse PWR using TRAC-PF1/MOD2 version 5.3.

Initially the uncertainties relevant to modelling the LBLOCA were identified and ranged using TRAC-PF1/MOD1. When TRAC-PF1/MOD2 and its documentation became available, most of the **important uncertainties** were implemented in it, namely: time in reactor life, fuel modelling, fuel conductivity, gas gap conductance, cladding conductivity, break flow, interphase drag, wall heat transfer (in part), minimum film boiling temperature, and effect of dissolved nitrogen in ECCS. Not yet implemented were: interphase heat transfer, wall heat transfer, MOD2 reflood model, some aspects of selection of scenario - primary pressure peaks, SGTR, and moderator feedback. Soon it was found out that the code version used had poor mass conservation, and for this reason only the first 50 s of the transient was studied.

A **base case** and **19 calculations** in which one uncertain quantity was varied at a time were run first. PCTs were calculated for two "hot rods", one with a total peaking factor of 1.7 (near the beginning of the reactor life, considered as reasonable upper limit), and the other with a peaking factor of 2.32 (design limit). It was found out that the time in reactor life/fuel state and the post-CHF heat transfer (post-blowdown only) had the **largest impacts** on the results. Medium impacts were recorded for: break flow (most significantly post-blowdown), bubbly-slug interphase drag (blowdown only), forced convection heat transfer and minimum film boiling temperature. Quantities with only small impacts were: cladding conductivity, entrained droplet size, entrainment fraction and stratified interphase drag.

Based on a straightforward examination of the above results, **two calculations** were made in which all the uncertainties were varied to maximise and minimise the PCT. To investigate possible nonmonotonic behaviour and/or synergisms **two additional single variations about both cases** were then made varying the break flow and the minimum film boiling temperature. More calculations would have been performed, if time would have permitted.

The **overall RURs** for PCT were: for the blowdown PCT, **737 - 1042 K** for the hot rod with a peaking factor of 1.7, and **909 - 1221 K** for the hottest rod (total peaking factor = 2.32). The post-blowdown PCT ranges were 548 - 1016 K and 601 - 1232 K for the two hot rods respectively. This study revealed no difficulties that cannot be overcome in practice and no problems that invalidate the AEA method. However, the calculations should be redone to cover the whole transient and additional possibly important uncertainties after the mass conservation problem has been cured.

The **discussions** within CSNI PWG2 have indicated that it is generally acknowledged that the AEA Method is well structured, is able to deal with both single- and continuous-valued parameters and may be able to generate useful information with less effort than the other two methods. However, it has been pointed out that the method generally relies heavily on experts' judgment, the number of necessary calculations is never known beforehand, and is based on the findings and judgment during the application, and, because some input parameters may produce nonmonotonic or even unexpected output variations, it is difficult to know for sure whether the right choices of input parameters and their variations to maximise/minimise the key output parameters were made. The method does not include any determination of the confidence level either. It has been explained that the customer is being convinced by presenting

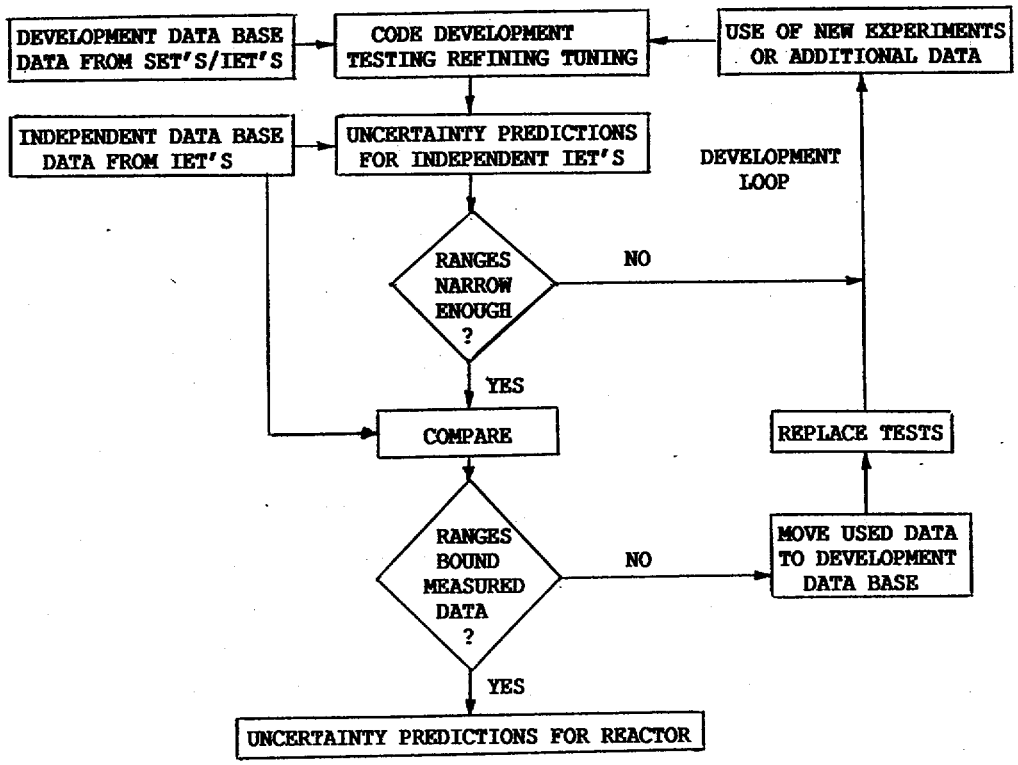

Figure 4 Use of Data Bases in AEA Method

sufficient evidence/reasons for the results.

5 STATUS OF RELATED FRENCH AND ITALIAN ACTIVITIES

In **France** the activities related to code uncertainty evaluation were essentially started in Grenoble in 1989 with the development and application of an "**Adjoint Sensitivity Method**" (ASM), which is a mathematical tool for obtaining the sensitivities of the calculated results to all parameters of interest in one separate computer run. The main limitation of ASM is its linearity. However, small variations can be considered also in a nonlinear system. Even in the case of "threshold" ("cliff edge") phenomena (like dryout) the sensitivity to the criteria of the phenomenon to occur can be addressed.

The method has been so far applied to two simple test cases, both involving a heated vertical pipe (CANON). The first case was purely hydraulic, but wall heat transfer was included in the second case. Comparisons of the calculated sensitivities to those obtained using most common methods have shown insignificant differences, although the computer time has decreased by an order of magnitude. These experiences indicate that ASM is potentially a very useful tool in ranking and selecting the most important input parameters for use in quantitative uncertainty amalyses. So far a module and a simplified version of CATHARE has been used in the applications, but work is going on to implement ASM in CATHARE 2 V1.4 as an integral part of the code. The complexity of the test cases is being increased.

After a thesis on ASM had been completed several new activities were initiated in France. A **new uncertainty evaluation method** is currently being developed taking into account experiences gained during the development and application of the other methods. Only the most important input parameters will be varied, as in the AEA method, but expert judgment will be avoided in their ranking and selection by use of ASM, and statistical methods will be used in conjunction with the propagation of their uncertainties through sensitivity calculations to generate the uncertainties of the desired output parameters, in a somewhat similar way as in the GRS method. **Assessment of the method itself** is also a part of the whole methodology, which includes feedback loops to code development as well. The French are prepared to use a considerable effort in order to develop tools which would reduce the required effort to apply the methodology in the long run.

In **Italy**, a **Fast Fourier Transform (FFT) Method** was developed at the Pisa University (PU) already several years ago and applied to ISP 21 (PIPER-ONE) and ISP22 (SPES). FFT is a method to quantify the accuracy (agreement) of a code in predicting measured quantities. The work at the PU has recently focused on the expansion of this method into an **uncertainty evaluation methodology** called UMAE (Uncertainty Methodology based on Accuracy Extrapolation). It attempts to focus on counterpart tests, and to arrive at the final uncertainties by extrapolating the

accuracies/uncertainties determined by calculating relevant experiments (using FFT) to full scale NPPs. Several conditions must be met before extrapolation may be performed, e.g. the accuracies should show an improving trend as a function of the scaling factor. So far the method has been applied to a PWR SBLOCA, and to a limited extent also to a BWR SBLOCA and PWR natural circulation. New applications and developments are under way.

The French and the Italians are also collaborating in an effort to apply and automatisize the Italian **Fast Fourier Transform (FFT) Method.** The first French-Italian application dealt with the pre- and posttest predictions of ISP27 (BETHSY), including the comparison of their results.

6 CONCLUSIONS

The activities related to quantitative thermal-hydraulic code uncertainty evaluation have been greatly **expanded** and **intensified** during the last couple of years. The oldest of them, CSAU, has been applied in several studies, the GRS and AEA Methods have undergone major development in conjunction with their first comprehensive testing/application efforts, and new related projects have been initiated in France and Italy.

One of the most important concerns with all of the methods is the **amount of effort** required to apply them. This question is being tackled by developing new mathematical tools, using the acquired experience and information in a systematical way, and by taking advantage of the fast progress in computer technology. It seems that considerable improvements are possible.

All of the method development teams have agreed that the time is ripe to hold a **Workshop early in 1994** to discuss the principals in detail, "align minds" and try to reach a common understanding on the underlying issues, while not avoiding controversial aspects. Maybe duplication of efforts can be reduced and cooperation further promoted.

The authors of this report agree that valuable increased knowledge of the accuracy and usefulness of all the methods could be gained by organising an **International Standard Problem** (ISP) and/or **Benchmark** (involving a large NPP), in which all the uncertainty methods would be applied and the results compared as is being planned in the framework of CSNI PWG2. One possibility would be to start with a simple test case, although it would not do justice to all the methods.

Also, it is recommended that the method development teams would repeat some of their application cases using **another version of the code** to study the relation between the resulting uncertainties and the code version.

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