REPORT OF A CSNI WORKSHOP ON
UNCERTAINTY ANALYSIS METHODS

London, 1-4 March 1994

Volume 2

August 1994
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

— to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
— to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
— to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium, Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971), New Zealand (29th May 1973) and Mexico (18th May 1994). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all European Member countries of OECD as well as Australia, Canada, Japan, Republic of Korea, Mexico and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

— encouraging harmonisation of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;
— assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;
— developing exchanges of scientific and technical information particularly through participation in common services;
— setting up international research and development programmes and joint undertakings.

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSONI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and coordinate the activities of the OECD Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of the nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

© OECD 1994

Applications for permission to reproduce or translate all or part of this publication should be made to:
Head of Publications Service, OECD
2, rue André-Pascal, 75775 PARIS CEDEX 16, France.
REPORT OF A CSNI WORKSHOP ON UNCERTAINTY ANALYSIS METHODS

A. J. Wickett
AEA Technology, Winfrith Technology Centre
G. Yadigaroglu
Swiss Federal Institute of Technology, Zurich

August 1994

SUMMARY

The OECD NEA CSNI Principal Working Group 2 (PWG2) Task Group on Thermal
Hydraulic System Behaviour (TGTHSB) has, in recent years, received presentations of a
variety of different methods to analyze the uncertainty in the calculations of advanced
unbiased ("best estimate") codes. At its meeting on 30 June - 2 July 1993 it accepted a UK
offer to host a Workshop to discuss the various methods that had been proposed. The PWG2
also asked that the Workshop discuss proposals for an International Standard Problem (ISP)
to compare the uncertainty analysis methods.

The objectives for the Workshop, defined at the December 1993 meeting of the TGTHSB
were:
- To discuss and fully understand the principles of uncertainty analysis relevant to
  LOCA modelling and like problems.
- To examine the underlying issues from first principles, in preference to comparing and
  contrasting the currently proposed methods.
- To reach consensus on the issues identified as far as possible while not avoiding the
  controversial aspects.
- To identify as clearly as possible unreconciled differences.
- To issue a "Status Report".

The Workshop was held on 1-4 March 1994 at the Strand Palace Hotel, London. A total of
30 delegates from 12 countries attended. Professor G. Yadigaroglu, of the Swiss Federal
Institute of Technology, Zurich, chaired the meeting on 1-3 March. Eight uncertainty
analysis methods were presented. A structured discussion of various aspects of uncertainty
analysis followed - the need for uncertainty analysis, identification and ranking of
uncertainties, characterisation, quantification and combination of uncertainties and
applications, resources and future developments. As a result, the objectives set out above
were, to a very large extent, achieved.

On 4 March plans for the ISP were discussed. Mr S. N. Aksan, Chairman of the TGTHSB,
chaired this part of the Workshop. The objectives for the ISP were redefined. A shortlist
of two experiments was drawn up.

Volume 1 of this report contains the record of the discussions and conclusions. Volume 2
contains accounts of the uncertainty analysis methods presented, provided by the authors.

Further copies of either volume of the report may be obtained from the OECD Secretariat or
from A. J. Wickett at the address on page i.

NEA/CSNI/R(94)20/Part 2 - iii -
<table>
<thead>
<tr>
<th>CONTENTS LIST</th>
<th>Page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Title Page</td>
<td>i</td>
</tr>
<tr>
<td>Summary</td>
<td>iii</td>
</tr>
<tr>
<td>Contents</td>
<td>v</td>
</tr>
<tr>
<td>G. E. Wilson, INEL</td>
<td></td>
</tr>
<tr>
<td>&quot;Overview of the CSAU Methodology for Quantifying Code Uncertainty&quot;</td>
<td></td>
</tr>
<tr>
<td>F. D'Auria, N. Debrecin and G. M. Galassi, Universities of Pisa and Zagreb</td>
<td></td>
</tr>
<tr>
<td>&quot;Outline of the Uncertainty Methodology based on Accuracy Extrapolation (UMAE)&quot;</td>
<td></td>
</tr>
<tr>
<td>M Leonardi, F. D'Auria, University of Pisa, and R. Pochard, CEA IPSN</td>
<td></td>
</tr>
<tr>
<td>&quot;The FFT based Method Utilisation in the Frame of the UMAE&quot;</td>
<td></td>
</tr>
<tr>
<td>S. N. Aksan, Paul Scherrer Institute, F. D'Auria and V. Faluomi, University of Pisa</td>
<td></td>
</tr>
<tr>
<td>&quot;Overview of UMAE-SETF (Extension at the Separate Effects Test Facilities of the Uncertainty Methodology based on Accuracy Extrapolation) and Qualification Process&quot;</td>
<td></td>
</tr>
<tr>
<td>A. J. Wickett, AEA Technology</td>
<td></td>
</tr>
<tr>
<td>&quot;A Review of LOCA Uncertainty Studies in AEA&quot;</td>
<td></td>
</tr>
<tr>
<td>AEA RS 5522</td>
<td></td>
</tr>
<tr>
<td>P. Lightfoot, Nuclear Electric and M. Trow, NNC</td>
<td></td>
</tr>
<tr>
<td>&quot;Development and Application of an Uncertainty Methodology for the Sizewell B Large LOCA Safety Case&quot;</td>
<td></td>
</tr>
<tr>
<td>H. Glaeser and E. Hofer, GRS</td>
<td></td>
</tr>
<tr>
<td>&quot;GRS Method (Germany)&quot;</td>
<td></td>
</tr>
<tr>
<td>H. Glaeser, E. Hofer, M. Kloos and T. Skorek</td>
<td></td>
</tr>
<tr>
<td>&quot;Uncertainty and Sensitivity Analysis of a Post-Experiment Calculation in Thermal Hydraulics&quot;</td>
<td></td>
</tr>
<tr>
<td>R. Pochard, A. Ounsi and A de Crécy, CEA</td>
<td></td>
</tr>
<tr>
<td>&quot;CEA/IPSN Uncertainty Method&quot;</td>
<td></td>
</tr>
<tr>
<td>E. J. Stubbe and L. Vanhoenacker, Belgatom</td>
<td></td>
</tr>
<tr>
<td>&quot;Methodology for Licensing of Non-LOCA Transients by means of the Code RELAP5/MOD2: Application: Feedwater Line Break&quot;</td>
<td></td>
</tr>
<tr>
<td>N. E. Lee and G. Menzel, ABB-CE</td>
<td></td>
</tr>
<tr>
<td>&quot;Limit Value Approach (LVA) for Uncertainty Evaluation of Best Estimate LOCA Analyses&quot;</td>
<td></td>
</tr>
</tbody>
</table>
OVERVIEW OF THE CSAU METHODOLOGY FOR QUANTIFYING CODE UNCERTAINTY

Presented at the OECD NEA CSNI Special Workshop on Uncertainty Analysis Methods, Strand Place Hotel, London, UK
March 1, 1994

Gary E. Wilson

INTRODUCTION

The Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology (Figure 1), and several applications, have been documented[1, 2, 3] and reviewed by the international community[4]. Therefore, the primary objective of this presentation is to provide the presenter's personal evaluation of the CSAU based on his experience in the original development and demonstration of the methodology, and subsequent applications.

The presentation is organized in three parts: a) review of the motivation and provisions for uncertainty analysis in the licensing of reactors in the United States (US), b) review of the objectives of the steps of the CSAU methodology in the context of advantages, disadvantages and potential improvement of the methodology based on recent experience, and c) a summary in the context of the objectives established for this meeting.

ROLE OF UNCERTAINTY ANALYSIS IN THE LICENSING OF US REACTORS

The motivation for uncertainty quantification may be considered twofold, although both objectives are tightly coupled:

1) To provide "best estimate plus uncertainty" analysis for licensing of commercial power reactors with realistic margins, as opposed to equally acceptable "evaluation model" analysis with typically conservative margins, and
2) To provide support to design and/or validation related analyses for research and production reactors.

The justification for "best estimate plus uncertainty" analysis is founded on 25 years of research[5] that has resulted in statutory provisions[6, 7] and guidance documentation[1, 2] directly related to its use. The basic structure for "best estimate plus uncertainty" analysis, pertinent to this presentation, is summarized in Figure 2.

REVIEW OF THE CSAU BASED ON APPLICATION EXPERIENCE

The developers of the methodology made a clear and separate distinction between: a) CSAU as a process, and b) any specific application of the process. As a process, CSAU is prescriptive regarding the elements to be considered in the uncertainty quantification, and the ultimate form of the uncertainty statement. However, at the process level CSAU is not prescriptive regarding application centered techniques, such as the statistical methodology to quantify uncertainty. For example, the applications related to the LBLOCA and SBLOCA demonstrations[1, 2] employed response surfaces coupled with Monte Carlo sampling to develop probability density functions (pdf) to describe the uncertainty in the safety parameters of interest. However, this approach to quantification of the uncertainty was at the discretion of the individuals performing the work, and was in no way a requirement of the CSAU process. Other comments to help clarify "required" versus "allowed" features of the CSAU will be provided in subsequent parts of this presentation.
Figure 1. Code scaling, applicability and uncertainty evaluation methodology flow diagram.
<table>
<thead>
<tr>
<th>Level</th>
<th>Source</th>
<th>Requirement</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>10CFR1.11</td>
<td>Protect public health and safety</td>
</tr>
<tr>
<td>2</td>
<td>10CFR100</td>
<td>Limit fission product release</td>
</tr>
<tr>
<td>3</td>
<td>Appendix A</td>
<td>Limit fuel failure</td>
</tr>
<tr>
<td>4</td>
<td>SRP 6.2 Containment</td>
<td>Limit RCS breach</td>
</tr>
<tr>
<td></td>
<td>SRP 15.1.4</td>
<td>Limit containment breach</td>
</tr>
<tr>
<td></td>
<td>TO 15.6.1 non-LOCA</td>
<td>RCS, Steam system pressure &amp; temp.</td>
</tr>
<tr>
<td></td>
<td>10CFR50.48</td>
<td>DNB (PWR), MCPR (BWR), Energy, Deposition, Fuel temp., Cladding strain</td>
</tr>
<tr>
<td>5</td>
<td>Reg. Guide 1.157</td>
<td>Peak cladding temperature, Oxidation, Hydrogen generation, Long term cooling, Coolable geometry</td>
</tr>
<tr>
<td></td>
<td>NUREG-1230</td>
<td>Establish code capability to analyze transients</td>
</tr>
<tr>
<td></td>
<td>NUREG/CR-5249</td>
<td>Determine code uncertainty</td>
</tr>
<tr>
<td></td>
<td>NUREG/CR-5616</td>
<td></td>
</tr>
</tbody>
</table>

Figure 2. Basis for "Best Estimate plus Uncertainty" licensing analysis in the US.
Steps 1 through 7 of the CSAU process establish:

a) The analysis envelop through the selection of the scenario(s) to be addressed (Step 1) and the nuclear power plant of interest (Step 2),

b) The required analysis capability (Step 3) in terms of the plausible phenomena of some significance, and the relative importance of those phenomena,

c) The actual analysis capability by selection of a computer code (Step 4), and determination of the code strengths and limitations (Step 5),

d) The boundaries within which to examine code induced elements of uncertainty (Step 6). The use of a code assessment matrix (Step 7) aids in defining these boundaries.

Based on his experience, it is the opinion of this presenter that the transient specific application required by the CSAU has:

a) An advantage because it results in the smallest uncertainty, with the strongest technical justification, for any one transient,

b) A disadvantage because it requires significant resources to cover all transients typically of interest in a licensing submittal,

c) The opportunity for improvement through two potential approaches to reduction of the required resources. That is, use of a "nucleus" set of transients with extensions to the complete analysis envelop by phenomenological similarity, and/or use of a "composite" PIRT and associated composite uncertainty determination.

Use of a "nucleus" set of transients is based on quantification of the uncertainty for each of a nucleus set of transients (for example SBLOCA, SGTR and MSLB), and demonstration by technical arguments that other transients are phenomenological subsets whose uncertainties are bounded by those of the nucleus transients. This approach will result in a significant reduction in the resources required to address the analysis envelop. However, it remains to be demonstrated that sufficient technical arguments can be developed to show the uncertainties of the nucleus transients fully bound that of the analysis envelop.

The composite PIRT approach is based on including the necessary parameters representing all transients of interest in one (perhaps two) PIRTs, ranging the parameters for all conditions represented by the PIRT, and then determining the uncertainties of interest for each composite PIRT. This approach has major advantages in the generality of the uncertainty quantification and the simplified analysis. A potential disadvantage is that the quantified uncertainty can be expected to be larger than for one transient treated individually. That is, there is some risk (considered small) the magnitude of the uncertainty will be unacceptable. It will also be necessary to define a way in which a single base case scenario can represent all transients of interest from which to range the input parameter variabilities. The concept underlying a composite PIRT is illustrated in Table 1.

For either of the two approaches just described, the use of the "Latin hyper-cube" method of statistical determination of the uncertainty is considered an attractive possibility for a more cost-effective analysis. Specifically attractive features of the methodology include:

a) The number of code calculations required is not a function of the number of input sources of uncertainty. Rather the number of required calculations is a function of the probability and confidence levels. Current information indicates 95/95% levels require 59 code calculations,

b) All sources of uncertainty are treated regardless of their relative importance,

c) The methodology (or equivalent) is being developed and demonstrated by others at this workshop.
<table>
<thead>
<tr>
<th>PHENOMENA</th>
<th>HIGHEST RANK¹</th>
<th>RELEVANT LOCATION</th>
<th>PREVALENT CONDITIONS²</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flashing</td>
<td>H</td>
<td>Cold legs</td>
<td>SBLOCA-III³ &amp; IV</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Core</td>
<td>SBLOCA-IV, MSLB-I</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Downcomer/lower plenum</td>
<td>SBLOCA-IV</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Hot legs</td>
<td>SBLOCA-II, MSLB-I</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Pressurizer</td>
<td>SBLOCA-I → IV, MSLB-I</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Steam generators</td>
<td>SBLOCA-IV, MSLB-I</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Upper head/upper plenum</td>
<td>SBLOCA-II → IV, MSLB-I</td>
</tr>
<tr>
<td>Heat transfer:</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ambient heat loss</td>
<td>M</td>
<td>Cold legs</td>
<td>SBLOCA-IV &amp; V</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Downcomer/lower plenum</td>
<td>SBLOCA-IV &amp; V</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Hot legs</td>
<td>SBLOCA-IV &amp; V</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Upper head/upper plenum</td>
<td>SBLOCA-IV &amp; V</td>
</tr>
<tr>
<td>Primary-to-secondary</td>
<td>H</td>
<td>Steam generators</td>
<td>SBLOCA-II, MSLB-I &amp; II</td>
</tr>
</tbody>
</table>

¹ - Guide to inclusion of phenomenon in uncertainty quantification
² - Represents summary designation that specifies required range of models/correlations associated with phenomenon of interest
³ - Represents summary designation that specifies transients and their time phases of interest

Table 1. Concept underlying use of a "Composite PIRT"
Based on his experience, it is the opinion of this presenter that the *frozen* code required by the CSAU has:

a) An advantage because it provides the "apples" plus "apples" uncertainty combinations that are needed in a licensing environment.

b) Has a disadvantage in that the end uncertainty may be larger than necessary, should a significant error be discovered, but left uncorrected during the uncertainty quantification.

It is this presenter's opinion the benefits of the advantage outweigh the restrictions of the disadvantage, given that a mature code is used in the uncertainty analysis.

Nodalization (Step 8) is treated as boundary condition in CSAU, rather than a source of uncertainty. That is, CSAU requires use of a "standard" nodalization that has been shown to adequately capture the dominant phenomena. The developers of the CSAU were not convinced that nodalization could be adequately treated in a probabilistic fashion, from both the technical and cost effective perspectives. The large range of nodalization variations result from the large number of NPPs, scenarios, and codes of interest, and user induced variability. The CSAU approach requires a sufficient body of information to establish an adequate standard nodalization. The determined uncertainty is then conditional on the standard nodalization. It is this presenter's belief that the technique works well with existing LWRs, and is still viable and remains advantageous for new reactor designs. This belief could be further validated by ISPs related to nodalization, user effects and the uncertainty methodologies.

Step 9 of the CSAU process characterizes and quantifies the individual uncertainties due to the experiments and codes. The experimental uncertainties arise from two sources: 1) Our limitations in a full understanding of "mother nature's" basic process, and 2) Limitations in our ability to accurately measure "mother nature's" processes. These sources of uncertainty were commonly treated in the CSAU demonstrations as the variability of physical parameters and coefficients. Examples include fuel density, thermal conductivity, and specific heat, gap conductance, etc., all treated with pdfs to describe their individual variability. The code induced uncertainties arise from limitations in our ability to model "mother nature" with absolute precision, modeling errors, and the compromises required to make the modeling tractable (i.e., run time constraints, smoothing needed at model interfaces, etc.). These sources were most often treated in the CSAU demonstrations as statistically defined variability, or by bounding biases (if the bias was large compared to the variability component of the uncertainty). Examples include heat transfer coefficients treated with pdfs, and coding errors in heat transfer coefficients treated with bounding biases.

Step 10 of the CSAU process characterizes and quantifies the uncertainty due to scale. This source of uncertainty arises because our understanding of "mother nature" is based on sample sets of limited geometric size. In the CSAU demonstrations these sources were most often treated with bounding variabilities based on extrapolation of variabilities at several subscale sizes, or with biases based on full scale data. Examples of the former include break flow and pump performance treated with bounding variabilities extrapolated from subscale data. An example of the latter is ECCS bypass during refill treated with a bounding bias (benefit) derived from full scale UPTF data.

Step 11 of the CSAU process characterizes and quantifies the uncertainty due to reactor input parameters and state. Uncertainties arising from reactor input parameters reflect our inability to precisely know and characterize geometric related elements. An example is the variability in fuel manufacture that results in a variability in core power. Uncertainties treated in this step also arise from our inability to know the exact reactor condition at the initiation of a transient; for example, actual reactor burnup history also resulting in a variability in core power. Uncertainties of this nature were most often treated in the CSAU demonstrations in one of three ways. If the variability was bounded by the variability introduced by related experimental induced uncertainty, the variability of interest was subsumed in the experimental variability. If the variability is not so bounded then the experimental variability was replaced by the larger variability of the reactor condition at accident initiation. Finally, if there was no related experimentally induced uncertainty then introduce a new
variability or bias as appropriate.

It is emphasized that, with one possible exception, the CSAU process does not prescribe how uncertainties will be determined. Rather, the CSAU process prescribes only that uncertainties arising from experiments, codes, scale and reactor input parameters and state will be determined in a technically justified and documented manner. The CSAU process may be interpreted as prescribing that a bias will be used appropriately when there is a true bias in the code treatment of a parameter of importance to determining uncertainty. The CSAU process allows the use of other biases if justified and documented.

The use of biases in the CSAU demonstrations has been perceived by some of the reactor safety analysis community as technically unjustified (or at least undesirable). In this presenter’s opinion, the allowance of bounding biases lends flexibility to quantifying uncertainty that, when treated in a conservative manner, justifiably recognizes the need for conservation of resources. The bounding biases must be introduced in a conservative form with regard to reactor safety decisions. The use of bounding biases recognizes that, on occasion, it is not cost effective to treat a source of uncertainty in a probabilistic manner. That is, the potential penalty in margin does not justify the cost of its reduction by treatment in a probabilistic fashion. This presenter considers the allowance of bounding biases as a strength of the CSAU process, not a limitation.

Steps 12 through 14 of the CSAU process are directed to the calculations and combinations of individual uncertainties and biases leading to determination of the total uncertainty associated with code simulations of reactor behavior. The CSAU process does not prescribe how the calculations and combinations will be performed, only that such analysis will be performed. The CSAU demonstrations used one possible approach to quantifying total uncertainty. That is:

a) Response surface representing dominant sources of uncertainty to replace the code to allow many samples in an economically acceptable manner,

b) Monte Carlo sampling to develop a total uncertainty pdf for those uncertainty sources treated in a probabilistic fashion, and

c) Algebraic addition of biases to the above pdf to accommodate penalties (or benefits) of uncertainty sources treated in a conservatively bounded manner.

In summary, this presenter has observed that the transition from mainframe computer to workstations has reduced the computer costs of for NPP calculations. The use of workstations has not significantly reduced the labor required to analyze the NPP calculations. Further reductions in the resource requirements appear to be possible with analysis methods that treat many scenarios as a single unit; for example the "nucleus" set of transients and composite PIRTs described earlier in this presentation. Use of probabilistic analysis methods that make the number of calculations independent of the number of sources of uncertainty (Latin hyper-cube) also show promise for reducing the cost of uncertainty analysis, and for increasing the quality of the uncertainty determinations.

SUMMARY OF THE CSAU METHODOLOGY IN THE CONTEXT OF THE OBJECTIVES OF THE WORKSHOP

The conclusion of this presentation is directed toward providing answers to the questions given in the instructions to the presenters.

Uncertainties are studied in the US because "Best estimate plus uncertainty" is an acceptable licensing approach leading to realistic margins with attendant improvement in safety and economics.

Sources of uncertainty addressed in the demonstrations of CSAU include: Experiments, Codes (including errors), Scaling (including compromises) and Reactor input parameters and state.
Sources of uncertainty are characterized in the *demonstrations* of CSAU as: Probabilistic in nature, True biases (may or may not include a probabilistic component) and Bounding biases adopted for economically justified reasons.

Quantification of uncertainties in the *demonstrations* of CSAU was based on:

- a) Identification of dominant sources of uncertainty using expert judgment founded on experimental evidence,
- b) Total uncertainty conditional on use of *standard* nodalization,
- c) Assignment of *real* pdfs to the variability of the sources of uncertainty where known,
- d) Assignment of *conservatively bounded, equally likely* pdfs to the variability of the sources of uncertainty where pdf poorly- or un-known, but well bounded,
- e) Assignment of *bounding biases* where true biases existed and the bias was >> probabilistic component of the variability,
- f) Assignment of *bounding biases* where the resulting penalty in margin did not warrant the resources required to use probabilistic methods,
- g) Extrapolation of *subscaling compromises* to full scale either as a probabilistic determine uncertainty or as a bounding bias.

Combination of uncertainties in the *demonstrations* of CSAU was based on:

- a) Response surface to represent code coupled with Monte Carlo sampling to determine pdf of total uncertainty arising from input parameter variabilities treated in a probabilistic manner,
- b) Algebraic addition of *bounding biases* to the pdf of total uncertainty,
- c) The developers of the CSAU methodology remain unconvinced that the way in which the method was applied required treatment of "cliff edges",
- d) Potential dependencies between sources of uncertainty were selectively treated through use of 1st, 2nd and 3rd order cross-products in the creation of the response surface,
- f) Scaling compromises were treated in a fashion similar to other sources of uncertainty.

CSAU has been used, to varying degrees, in at least six applications:

- a) Fully applied in two studies related to LBLOCA and SBLOCA,
- b) Methods that are similar have been applied by GE and WEC,
- c) Valuable insights regarding potential improvements in the methodology have been gained from partial applications of the methodology in several other studies related to establishing system code and experimental program requirements.

Factors related to required resources include:

- a) The development and initial demonstration of CSAU required approximately 9 and 4 man-years, respectively (single NPP, single scenario application),
- b) Without further improvement, it is likely 2 to 3 man-years can be expected to be required for any subsequent similar application,
- c) This presenter believes that the potential methodology improvements already mentioned (composite PIRT and Latin hyper-cube sampling) hold promise for significant further reductions in required resources.

Relative to future developments and/or applications:

- a) The USNRC and its contractors (INEL, BNL, LANL, SNL) are using CSAU in the ALWR certification program to:
  
  1) Establish the code requirements of the system analysis code(s) that will be used to audit the vendor certification submittals,
  
  2) Develop the USNRC experiments that will provide the assessment data for the above code(s),
  
  3) Determine the uncertainty in the above code(s),

8 of 9
b) This presenter believes that methodology improvements will be a natural consequence of the above effort possibly along the lines already suggested,

c) There is evidence that the vendors are using analytical methods for uncertainty quantification that are compatible with the "best estimate plus uncertainty" provisions of the existing licensing requirements of the USNRC.

REFERENCES


OUTLINE OF THE UNCERTAINTY METHODOLOGY
BASED ON ACCURACY EXTRAPOLATION (UMAE)

F. D'Auria (*)
N. Debrecin (*)
G.M. Galassi (*)

(*) University of Pisa – Via Diotisalvi, 2 – 56100 Pisa – Italy
(*) University of Zagreb – Croatia

Special Workshop on Uncertainty Analysis Methods

London (UK), March 1–3, 1994
OUTLINE OF THE UNCERTAINTY METHODOLOGY BASED ON ACCURACY EXTRAPOLATION (UMAE)

F. D'Auria(\*), N. Debrecin(\+), G.M. Galassi(\*)

(\*) University of Pisa - Via Diotisalvi 2 - 56100 Pisa - Italy
(\+) University of Zagreb - Croatia

Abstract

The present paper deals with the description of the UMAE (Uncertainty Methodology based on Extrapolation of Accuracy) methodology suitable to evaluate the uncertainty in the prediction, by thermal-hydraulic system codes, of the transient scenarios in nuclear reactors.

The methodology is based upon the extrapolation of the accuracy resulting from the comparison between code results and relevant experimental data obtained in small scale facilities.

A simplified logic diagram of the UMAE is compared with a similar one derived for CSAU (Code Scaling, Applicability and Uncertainty evaluation) previously proposed by United States Nuclear Regulatory Commission.

A few results related to the full application of the UMAE to a small break Loss of Coolant Accident in a PWR, including core uncoverage, are also reported.

1. INTRODUCTION

Evaluating nuclear power plant performance during transient conditions has been the main issue of safety researches in the thermal-hydraulic area carried out all over the world since the beginning of the exploitation of nuclear energy for producing electricity in the 50's (e.g. State of the Art Report by CSNI and Compendium of ECCS Researches by US NRC, both issued in 1989, refs. [1] and [2]).

A huge amount of experimental data has been made available from very simple loops (Basic Test Facilities and Separate Effect Test Facilities) and from very complex Integral Test Facilities simulating all the relevant parts of a Light Water Reactor. On the other hand, sophisticated computer codes like Athlet, CATHARE, RELAP and TRAC have been developed in Europe and United States and are widely in use at present. These are able to calculate time trends of any interesting quantity during any transient in LWRs with assigned boundary and initial conditions. The reliability of the predictions cannot be directly assessed owing to the lack of suitable measurements in the plants. So, the capabilities of the codes can be evaluated only from the comparison between calculation results and experimental data recorded in small scale facilities. In order to evaluate the applicability of a code in predicting a plant situation, one must be sure, at least, that the experimental data used for qualifying the codes are representative of phenomena expected in the plant and, subsequently that codes are able to reproduce qualitatively and quantitatively these data.

The (unknown) error made in predicting plant behaviour is called uncertainty, while the discrepancies between measured and calculated trends related to experimental facilities, are included in the accuracy of the prediction. The methodology here proposed aims at evaluating uncertainty from accuracy data.

The purposes of this paper are to give an outline of the UMAE (Uncertainty Methodology based on Accuracy Extrapolation), to compare it with the CSAU methodology (Code Scaling, Applicability and Uncertainty evaluation, e.g. refs. [3], to [6]), and to show few results of its application to a small break LOCA in a typical PWR.

2. BASIS AND DEVELOPMENT OF THE METHODOLOGY

The fundamentals of the methodology have been discussed in previous papers (e.g. refs. [7] to [11]) and can be drawn from the considerations 1 to 6 below, related to the test facilities and the codes. It is assumed that both of these are representative of the actual state of the art: i.e. facilities are "simulators" of a reference LWR and codes are based on the "six balance equations model" and must be
intended as generically "qualified", ref. [12].

1. The direct extrapolation of experimental data is not feasible; nevertheless, the time trends of significant variables measured during counterpart tests in differently scaled facilities are quite similar: this fact must be exploited.

2. Phenomena and transient scenarios occurring in larger facilities, being nearly constant the other conditions (e.g. design criteria, quality of instrumentation, etc.), are more close to plant conditions than those recorded from smaller facilities.

3. "Qualified" codes are indispensable tools to predict plant behaviour during nominal and off-nominal conditions.

4. The confidence in predicting a given phenomenon by the code, must increase when increasing the number of experiments analyzed dealing with that phenomenon.

5. The uncertainty in the prediction of plant behaviour cannot be smaller than the accuracy resulting from the comparison between measured and calculated trends; furthermore, accuracy (uncertainty) must be connected with the complexity of the facility (plant) and of the considered transient.

6. The effects of user and of nodalization must be included in the methodology.

The basic idea is to get the uncertainty from considering the accuracy. The main problems to achieve this are connected with the availability of experimental data that must be 'representative' of plants, with the quantification of accuracy and with the justification of any relationship between accuracy obtained in small dimensions loops with accuracy of the plant calculation, i.e. uncertainty.

The use of data base from counterpart and similar tests in Integral Test Facilities was of large help in this context, with main reference to the last mentioned problem. In particular, similar tests are those experiments performed in differently scaled facilities that are characterized by the occurrence of the same thermalhydraulic phenomena; counterpart tests are similar tests where boundary and initial conditions are imposed following a scaling analysis, ref. [9].

Two main aspects have been considered at a preliminary level to judge the realism of the accuracy extrapolation: from the experimental side, the design of the facilities, the boundary and initial conditions of the experiments, the suitability of the instrumentation, the quality of the recorded data have been evaluated together with the similarity of the phenomena (e.g. ref. [13]); from the code side, the general qualification process, the capability to simulate the relevant phenomena identified, the qualification of the nodalization and of the code user have been independently assessed (e.g. refs. [14] and [15]).

2.1 Extrapolation of accuracy

Several parameters of geometric or thermohydraulic nature can be used, in principle, to derive uncertainty from accuracy values; furthermore, the use of a unique parameter is preferable to minimize the possibility of counterfeiting information from the available data base.

The selected parameter should constitute a link between the experimental data and the data foreseeable in the plant in the case of occurrence of the conditions of interest, it must be representative of the involved phenomena at a global scale (i.e. the equivalent diameter in the core simulators is representative of thermalhydraulic phenomena in the core but does not affect substantially pressure behaviour during a depressurization transient), it must take into account the present status of the technology (i.e. the height of the facilities or the initial pressure might not be suitable parameters because they are nearly the same in the different facilities and in the reference plant).

Several parameters fulfill the above requirements, ref. [16], nevertheless the volume of the facility or, in dimensionless form, the ratio between primary system fluid volumes in the facilities and in the reference plant was selected as 'extrapolation' parameter.

The situations depicted in Fig. 1 are expected from using such an approach (e.g. ref. [10]). The volume scaling ratio is reported on the horizontal axis; this quantity actually varies by four orders of magnitude from the smallest facility to the plant (e.g. refs. [12] and [17]). The shaded area represents the range of facilities, the smallest one and the largest one are characterized by a volume roughly 2000 times and 50 times, respectively, lower than that of a 1000 MWe plant. On the vertical axis the generic quantity \( Y \) and the ratio \( YE/YC \) (experimental over calculated value) are represented: \( Y \) is the value of any quantity that is relevant in a given transient such as Peak Cladding Temperature, Time of DNB, minimum level in the core, time of actuation of any safety system. The ratio \( YE/YC \) measures the accuracy of the code calculation in predicting the quantities mentioned. The same ratio can be substituted by a quantity which gives the measurement of the overall accuracy of the code results in predicting the same scenario (e.g. utilizing a specific methodology based on the Fast Fourier Transform of the experimental and calculated trends, refs. [18] and [19]).
For both cases (a) (Y versus \( Y_C \)) and (b) (\( Y_E/Y_C \) versus \( Y_C \)), three situations may occur from correlating the data in Fig. 1, where the value 1 in the vertical axis represents the maximum achievable accuracy (\( Y_E/Y_C \) or a reference value related to the prototype:

- curve 1: the measured values tend toward a value that is typical of the plant and the calculation accuracy improves when the dimensions of the facility increase;
- curve 2: the opposite is true: measured values go away from the reference value, and calculation accuracy gets worse and worse when the dimension of the facility increase;
- curve 3: the data are randomly dispersed around the reference value or the unity value.

Case 1 is obviously preferable for drawing conclusions from both approaches (a) and (b); case 2 prevents any scaling possibility of the considered data; case 3 does not permit us to draw any conclusion about scaling. Situation 3 was retained the most probable before starting the analyses, ref. [9].

The construction of large scale facilities (i.e. Bethsy and Lstf, ref. [17]) and the availability of data from the related researches, justifies this approach by assuming that the increase in geometrical dimensions leads to the reduction of the unavoidable scaling distortions like those associated to the heat release from structures during a depressurization transient.

The data to be included in diagrams like that depicted in Fig. 1, must be representative of a unique phenomenon, in order to make the extrapolation meaningful: the error (characterized by the ratio \( Y_E/Y_C \)) in predicting the pressure during a fast depressurization like that caused by a large break LOCA should not be combined with the error in predicting the same quantity during a small break LOCA when different phenomena and aspects, including nodalization detail, affect the prediction itself. Data from counterpart and similar tests are useful in this connection, too.

2.2 Exploitation of counterpart test data

The achievement of counterpart tests data implies the choice of a reference experiment, the design of the boundary and initial conditions and, clearly, the availability of proper facilities. The complexity both in the hardware and in the management of the facilities and their inherent peculiarities make a huge amount of resources necessary for performing optimal counterpart tests: notwithstanding this and the availability of substantially agreed criteria for the definition of test specifications, ref. [9], the actual data bases of counterpart and similar tests suffer of practical limitations outlined in refs. [20], [21] and [22]; these are related to counterpart or similar test activities, small break LOCA in BWR, natural circulation in PWR and Loss of Feedwater in PWR, respectively.

Recently the widest counterpart test activity has been carried out with reference to a small break LOCA in PWR utilizing four facilities: Lobi, Spes, Bethsy and Lstf (refs. [23] and [24]). In each of the two smallest facilities, Lobi and Spes, two experiments have been performed which differentiate for the initial power level.

The steps necessary for the achievement and the exploitation of the counterpart tests data can be drawn from the diagram in Fig. 2. The right hand side and the left hand side of the diagram should be distinguished, dealing with code related activities and analysis of the experimental data, respectively. Both of these contribute at obtaining the main goal that is, block nr. 14, the definition of the reference plant scenario in case of the considered transient.

The choice of a transient is included in the block nr. 1, that deals with the selection of suitable boundary and initial conditions. Considering the peculiarities of each facility, ref. [17], and the thermohydraulic scaling laws, e.g. refs. [25] and [26], in blocks nrs. 2 and 3 respectively, counterpart test criteria can be used to get detailed test specifications in each facility, block nr. 4. Pre-test calculations can be performed, eventually to confirm the validity of the specifications, block nr 7. Examples of the use of such a procedure can be found in refs. [27] and [28] related to the design of counterpart experiments in Piper-one and Spes facilities, BWR and PWR simulators, respectively.

The execution of the experiments, block nr. 5, and especially the comparison among the actual boundary and initial conditions and the measured trends in the different facilities, block nr. 6, leads eventually to the decision to consider the tests as counterpart or similar experiments. Criteria to arrive at this decision are discussed in the next section (see also ref. [26] and [29]).

In principle, experimental data can be extrapolated to get a reference plant behaviour, block nr. 14, but this proved to be unrealistic without the support of independent analytical tools; this can be seen from considering the results of refs. [13] and [30].

So, codes are strictly necessary to obtain qualified plant scenarios. Qualified codes, e.g. ref. [15], and qualified nodalizations, ref. [31], set-up and used by qualified users, e.g. refs. [32] and [33], block nrs. 8 to 10, are necessary to perform post-test analysis of each experiment also to evaluate the capability of the code to calculate the same phenomena at different scale, block nr. 11. The comparison of experimental data included in this block, allows the optimization of the facility nodalizations, e.g. ref. [34]: this experience should be transferred when developing the plant nodalization, block nr. 12.
The comparison between the results of plant calculations and the data measured or calculated in the various facilities with reference to the same experiment, block nr. 13, leads to the Analytical Simulation Model, block nr. 14, that can be used to perform the reference plant calculation; the resulting trends constitute the basis for calculating the uncertainty as shown in the next sections.

3. OUTLINE OF THE UMAE

The use of calculated and measured data related to counterpart and similar tests, especially with the help of the code, directly led to the achievement of quantities connected with the uncertainty. "Dispersion bands" and "extrapolated" plant behaviour were defined and evaluated in refs. [7], [8], [20] to [22] and [35]; in one case, single phase natural circulation in PWR, ref. [21], an analytical relationship was established and tested between the interesting quantity (core flowrate) and the volume scaling ratios of the involved facilities.

The UMAE procedure aims at calculating the uncertainty and, although making use of the same above-mentioned data base, involves the rationalization of the various steps including the use of statistics in order to avoid or minimize the expert judgment at the various levels. Generic experiments in integral facilities and related calculations other than counterpart and similar tests, can be processed by the UMAE, provided the availability of data base related to a reasonable set of individual phenomena that envelope the key phenomena foreseeable in the selected plant scenario.

A simplified flow-diagram of the UMAE is reported in Fig. 3: the blocks are identified by lower letters and the relevant connections by capital letters. The following logical steps or conditions to be verified are part of the methodology (some of these, mentioned below, are quite obvious and are not included in the figure):

- **'Frozen' Code** (block "a"): an internationally recognized code version must be available; the consequences of the installation of the code on the computer must be checked, e.g. refs. [36] and [37].
- **Reactor and Accident Scenario**: a PWR or a BWR can be selected; the choice of the reactor system and of the accident scenario should be restricted to those for which valuable assessment activity has been carried out, e.g. ref. [1].
- **Relevant Experimental Data**: experimental data include hardware data of facilities, suitable boundary and initial conditions and time trends of important quantities measured during the experiments of interest.
- **Code and User Capabilities** (block "b"): the code must be the object of a wide international use; no special deficiencies should have been detected in predicting the phenomena to be considered, e.g. refs. [1], [2] and [15]. The code user or group of users should be properly qualified: the problems connected with code user and the meaning of user qualification can be found in refs. [33] and [14], respectively.
- **Suitability of Integral Facilities Design**: the scaling and design factors of the involved test facilities must comply with the actual state of the art in the field, refs. [26] and [38]. This includes unavoidable scaling distortions like the heat release from structures that must be carefully evaluated. In relation to scaling, the facilities considered in ref. [17] appear acceptable.
- **Suitability of Test Design**: a scaling analysis must be performed to fix boundary and initial conditions especially in the case of counterpart tests; reactor conditions should be taken as reference in each case. Criteria in ref. [9] can be considered.
- **Suitability of Test Data**: instrumentation, data acquisition system and assumptions made to derive complex quantities like total mass in primary system, heat losses to environment, etc., should be checked.
- **Development of Nodalizations** (blocks "c" and "i"): a qualified user (or group of users), in the sense defined in ref. [14], must build up the input data decks of the involved facilities and plant. In the case of Relap5 code, specifications in refs. [39] and [40] should be followed.
- **Generic' and 'Specific' Experimental Data** (blocks "d" and "h"): 'generic' and 'specific' experimental data are derived from the involved facilities; the former set of data is not necessary in the accuracy extrapolation process and must be used for the independent qualification of the nodalizations (an example of this can be found in refs. [15] and [41]); the latter must include all the key phenomena expected to occur during the considered transient.
- **Nodalization Qualification** (block "g"): the developed nodalization must be qualified considering the comparison with hardware data, boundary and initial conditions and time trends of relevant quantities. A procedure has been developed and applied in which a 'steady state' and a 'transient' level of qualification are distinguished (e.g. refs. [32] and [40]); criteria for the selection of relevant quantities are discussed below.
Evaluation of the Specific Data Base: The specific data base is constituted by the signals recorded during the considered experiments and by the results of the code calculations. Each test scenario (measured or calculated) should be divided into 'Phenomenological Windows' (Ph.W), ref. [1]. In each Ph.W, 'Key Phenomena' (K.Ph) and 'Relevant Thermallydraulic Aspects' (RTA) must be identified. K.Ph characterizes the different classes (e.g. small break LOCA, large break LOCA, etc.) of transients and RTA are specific of the assigned one; K.Ph as defined in ref. [4] are always applicable; an example of definition of RTA for small break LOCA in PWR can be found in ref. [24]. K.Ph and RTA qualitatively identify the assigned transient; in order to get quantitative information, each RTA must be characterized by 'Single Valued Parameters' (SVP, e.g. minimum level in the core), 'Non-Dimensional Parameters' (NDP, e.g. Froude number in hot leg at the beginning of reflux condensation), 'Time Sequence of Events' (TSE, e.g. time when dryout occurs) and 'Integral Parameters' (IPA, e.g. integral or average value of break flowrate during subcooled blowdown). Example of derivation of SVP, NDP, TSE and IPA can be found in ref. [24].

Similarity of Experimental Data (path "FG"): RTA, SVP, NDP, IPA and TSE are used in each Ph.W to demonstrate the similarity among the available experimental data, ref. [24]. If this is not achieved, UMAE cannot be used.

Acceptability of Calculation Results (block "e"): RTA, SVP, NDP, IPA and TSE are used in each Ph.W to demonstrate the qualitative accuracy of the calculation, refs. [34] and [43], (again path "FG" in Fig. 3). If this is not achieved UMAE cannot be used.

Accuracy Quantification (block "f"): if the above two steps are acceptable, the accuracy of code calculations can be quantified utilizing the already mentioned special procedure (refs. [18] and [19]). This produces a single value in the frequency domain from each comparison between calculated and measured transient scenario; the so-called 'Average Accuracy' must be smaller than an assigned value, ref. [44].

Feedback on the Plant Nodalization (path "GI"): the result of block "g" is a set of qualified nodalizations that predict satisfactorily the assigned transient or the related K.Ph if counterpart tests are not available. The relevant experience gained in this process must be transferred for setting up the plant nodalization.

Plant Calculation (block "j"): two plant calculations must be carried out: a) the 'facility K_V scaled' and b) the 'realistic conditions' calculation. In the former case boundary and initial conditions utilized as input should be derived from those in the experimental facilities following the counterpart test scaling criteria in the ref. [9]. In the latter case nominal conditions should be used.

Acceptability of Plant Calculation (block "k"): RTA, SVP, NDP, IPA and TSE are used in each Ph.W to demonstrate the similarity of the plant predicted scenarios in the 'facility K_V scaled' case with measured and calculated scenarios in the facilities. Furthermore, Ph.W, and K.Ph must be the same in the cases a) and b) of the previous step. If these conditions are not fulfilled, UMAE cannot be used.

Analytical Simulation Model, ASM (block "m"): this essentially consists of a 'qualified' plant nodalization running on a 'qualified' code by a 'qualified' user. The ASM can be used to predict plant scenarios characterized by the same Ph.W and K.Ph as the assigned transient.

Accuracy Extrapolation (block "l"): summing up, if the following conditions are fulfilled, accuracy in predicting SVP, NDP, IPA and TSE can be extrapolated, ref. [45]:
- the design scaling factors of the involved facilities are suitable;
- the test design scaling factors of the involved experiments are suitable;
- the experimental data base is qualified;
- the nodalizations and the related users are qualified;
- RTA are the same in the considered experiments if counterpart or similar tests are involved; otherwise, the same RTA can be identified in different experiments;
- RTA are well predicted by the code at a qualitative and a quantitative level;
- RTA are the same in the plant calculation 'facility K_V scaled' and in the experiments; parameters ranges (SVP, NDP, TSE and IPA), properly scaled, are also the same. This must be interpreted in different ways depending upon the availability of counterpart tests;
- in the plant calculation 'realistic conditions' Ph.W and K.Ph are the same as in the considered experiments; SVP, NDP, TSE and IPA may be different: reasons for this are understood.

The extrapolation of accuracy is achieved with reference to the above mentioned parameters through the use of the statistics, refs. [43] and [46]. The ratios of measured and calculated values of SVP, NDP, TSE and IPA are reported in diagrams like that shown in Fig. 1 assuming that they are randomly distributed around the unity value. This is also justified by the huge number of variables affecting the considered ratios. In this way 'mean accuracy' and '95th
percentile accuracy’ are derived that are applicable to the plant calculation. In this procedure the measurement errors, the unavoidable scaling distortions and the dimensions of the facility are directly considered.

* Uncertainty Calculation (block "n"): the extrapolated accuracy can be superimposed directly to the results of the ASM calculation. Nevertheless, some additional analytical elaboration is necessary to transform the point values into continuous error bars that envelope the reference ASM calculation, ref. [47]. In particular, ‘timing’, ‘spatial’ and ‘mean integral’ uncertainty have been defined, taking as reference TSE, SVP-NDP and IPA values, respectively.

3.1 Differences with respect to CSAU

Other than CSAU, methodologies to evaluate uncertainty are proposed by different European organizations, ref. [48]. So far only CSAU has been fully applied to get uncertainty of code calculations, e.g. refs. [49] and [50].

A simplified flowsheet of CSAU is given in Fig. 4 (more details can be found in refs. [3] to [6]). In both Figs. 3 and 4, the dashed blocks represent steps or conditions that are common to CSAU and UMAE. Although the approach may be different, these essentially require the same kind of analysis in both cases and will not be discussed further; i.e. the code applicability, block "b" in Fig. 4, can be found in block "b" and partly in block "h" of Fig. 3.

Three main differences between UMAE and CSAU can be identified:

a) only expert (or engineering) judgment can stop the process of getting uncertainty in the case of CSAU (blocks "e", "f" and "I" in Fig. 4), while a detailed comparison between measured and calculated trends may arrive at the same result for UMAE (path "FG" in Fig. 3);

b) several sensitivity calculations using a plant nodalization essentially approved by expert judgments are foreseen in the CSAU to get uncertainty; one plant calculation through a qualified nodalization is foreseen in the UMAE;

c) in order to get uncertainty from UMAE, experimental data in integral test facilities must be available and related to the assigned transient; this is not the case in CSAU. Furthermore the applied code must be able to predict the measured scenario.

Minor differences between UMAE and CSAU are connected with:

- the user qualification: an unqualified user, presumably will not get acceptable results from the block "F" in Fig. 3, while he can perform sensitivity calculations in the CSAU;
- errors that can be present in the plant nodalizations of both CSAU and UMAE: the probability that this happens in the UMAE is minimized due to the analysis at block "k" in Fig. 3;
- the use of the response surface methodology is included in CSAU and is not in the UMAE;
- the assumption in the UMAE that $\tan \phi$ / $\gamma_c$ is a statistical quantity.

The consequences of the identified differences between UMAE and CSAU can be outlined by the study a Loss of Feedwater scenario in PWR, refs. [22] and [51]. The considered experimental scenario is characterized by the full loss of feedwater, no actuation of high pressure bleed systems in primary side and delayed actuation of emergency feedwater in steam generators following the occurrence of core dryout. The actuation of emergency feedwater in the voided steam generators causes, when it reaches the bottom of the steam generator tubes, void collapse in the primary side and draining of the liquid stored at that moment in the pressurizer owing to the pressurizer PORY cycling. The liquid drains toward the core and leads to quench in a situation of primary system pressure decrease. Code calculations were performed, e.g. ref. [51]. When considering the process along the path "FG" in Fig. 3, it was evident that the prediction of the RTA associated with the heat transfer from steam generator walls to the falling emergency feedwater was not adequate and led to potential errors in predicting rod surface temperature larger than 500 K (this is not acceptable because the limits of cladding damage are reached). In this situation, the code Relaps5/mod2 could not be used for plant calculation in the UMAE. There is no brake for the use of this same code in the CSAU to predict plant behaviour in the case of the considered transient.

Another example from which the differences between UMAE and CSAU arise, lies in the choice of the code. A simple code that has been the object of wide assessment (e.g. Relap4/mod6) can be used, in principle, by the CSAU and can produce uncertainty values, possibly lower than those calculated by a 'sophisticated' code. In the UMAE, the use of such a code is prevented, if the case, again by the logic intrinsic into the path 'FG' in Fig. 3.

4. APPLICATION OF UMAE

As already mentioned, parts or concepts of the UMAE have been applied to transient scenarios like small break LOCA in BWR (e.g. ref. [20]), natural circulation in PWR (e.g. ref. [21]). Loss of
feedwater in PWR (e.g. ref. [22]); in these cases, quantities connected with uncertainty have been derived.

The only complete application, so far, has been carried out with reference to a small break LOCA in PWR (e.g. ref. [23]), where LoBi, Spes, Bethsy and Lstf are involved. Significant results from this analysis are summarized hereafter.

The transient is originated by a side oriented break in the cold leg having an area equivalent, roughly to 6% of the area of the cold leg itself. The sequence of interventions is typical of this kind of transient in a plant (Tab. 1): after the break occurrence, scram and pumps trip are provided together with a signal for isolating steam generators. Accumulators intervention is foreseen when primary pressure falls below 4.2 MPa. Following accumulators emptying, break flowrate causes mass depletion in the primary system leading unavoidably to core dryout toward the end of the transient. In LoBi and Spes, Low Pressure Injection System actuation is foreseen and is effective in rewetting the rods. An idea of the transient scenario can be drawn from Figs. 5 to 8; the first dryout situation at about 100 s into the transient should be noted: this is quenched by loop seal clearing. The transient was subdivided into four Ph.W (ref. [13]) and more than 40 RTA were identified. Some experimental values of TSE and SVP are given in Tab. 2.

The results shown in ref. [34] demonstrate that the Relap5/Mod2 code is able to predict all of these. Few values of TSE and SVP in the form $Y_F/Y_C$ are shown in Figs. 9 to 12.

The application of the statistical methodology, ref. [46], dealing with step "I" in Fig. 3, led to the 'average' and '95th percentile' uncertainty values shown in Tab. 3, in relation to a few of TSE, SVP, NDP and IPA.

The last step (block "n" in Fig. 3) consisted in deriving continuous values of 'timings', 'spatial' and 'mean integral' uncertainty, ref. [47], to be superimposed to the ASM calculation results carried out in 'realistic conditions' (ref. [30]). The Westinghouse 630 Mwe two loops plant of Kırko (Slovenia) was selected as reference reactor for the ASM calculation.

The results related to the rod surface temperature trend and to the fluid mass lost from the break are shown in Figs. 13 and 14, respectively. For simplicity only the envelope of the curves obtained from the three above-mentioned types of uncertainty are shown. It can be noted that the positive error in predicting the peak cladding temperature is of the order of 100 K. The occurrence of the first dryout situation combines, in the 'realistic conditions' calculation with the second one occurring in the experiments.

4.1 Additional biases to be considered

A list of additional biases to be considered in the UMAE is given in Tab. 4. Four possible sources of additional uncertainty are identified and the related applicability to the general case and to the present case is distinguished.

Additional user effects must be considered in the cases when the 'realistic conditions' calculation needs the modeling of systems or components not included in the 'facility $K_V$ scaled' calculation. A similar situation may occur when a component, important in the assigned transient cannot be properly scaled, i.e. pump, dryer, etc. This does not happen in the present case.

In general the overall nodalization scheme of the plant may not reflect entirely the nodalizations adopted in the scaled facilities, although criteria for nodalization development are pursued. A typical example of this is represented by the need, in some situations, to introduce parallel channels in the core or in the downcomer regions to simulate "different one-dimensional" flowpaths, i.e. hot channel and hot rod behaviour in the core. Again, sensitivity studies focused to assess the differences between the potential ASM and the facilities nodalizations must be carried out.

Important multidimensional effects may be hypothesized during the course of the assigned transient in the plant, but could be not present or of less importance in the test facilities, refs. [52] to [54]. Typical examples are represented by natural circulation flows inside the core, ref. [54], by CCFL (CounterCurrent Flow Limitation) breakdown in some areas of the core upper tie plate, refs. [53] and [54] and by the occurrence of reverse flow in steam generator tubes, e.g. ref. [55]. If these phenomena are expected to be important and only 1D codes are available, specific changes to the nodalization should be introduced that should be qualified through the comparison with experimental data recorded in the relevant facilities, refs. [52] to [54].

Nuclear rods are not available in most experimental facilities: modeling of this may be the source of additional uncertainty. Nevertheless, the impact of this on the ASM calculation results can be easily accounted for by specific sensitivity calculations by varying, especially, input values in the nodalization like gap or UO₂ conductivity and gap thickness.
5. CONCLUSIONS

The UMAE methodology has been presented that allows the calculation of uncertainty in the predictions of off-normal conditions in nuclear reactors by thermalhydraulic system codes. Essentially, the uncertainty is derived from the extrapolation of accuracy obtained by comparing measured and calculated variables trends that are relevant to the assigned accident scenario. The use of data from counterpart tests, was of large help in developing the concepts at the basis of the methodology.

One complete example of application of the UMAE to a small break LOCA in PWR has been given. In that case dryout is foreseen and peak cladding temperature, including the error of the order of 100 K, remains well below the licensing limits.

The main peculiarity of the UMAE is the direct exploitation of the huge experimental data base now available from integral test facilities: this is strictly necessary and is retained suitable to characterize any foreseeable accident scenario in Light Water Reactors. Moreover, all the aspects contributing to the uncertainty are taken into consideration: the user effect, the nodalization qualification, the errors in introducing boundary and initial conditions, the intrinsic capabilities of the code in predicting the phenomena of interest, are fully included in the final evaluation of the error bands.

The comparison with the CSAU methodology demonstrated that the expert judgment is avoided or minimized in the UMAE. Furthermore, the logic of UMAE prevents its use when phenomena expected in the plant are not satisfactorily simulated by the code or when the code itself is based upon too simplified assumptions.

Few additional biases to be included in the final value of uncertainty that are not fully considered by the UMAE, have been identified. Among these, the potential occurrence of important multidimensional phenomena represents a critical issue if only codes based on one-dimensional models are available: experimental data from specific facilities should be used to ascertain the codes capabilities in this connection.

Finally, it can be mentioned that the UMAE development required an effort equivalent, roughly to 10 man-years; the resources necessary for its use are dependent upon the availability or less of qualified nodalizations and experimental data and can be estimated in the range 1-3 man-years to get acquainted to its use and to calculate uncertainty related to any assigned transient. Lower resources are needed for the standard use.

ACKNOWLEDGEMENTS

More than fifteen graduate, undergraduate and Ph.d students, researchers and professors at Universities of Pisa and Zagreb, took part to the development of UMAE and to its qualification. Almost all of them are coauthors of the papers listed in the references. Their contribution has been invaluable.

One of the authors (F. D'Auria) wishes to acknowledge the contribution of all the members of the CSNI THSB Task Group for the discussions related to this subject had in different occasions.

REFERENCES

/1/ CSNI Group of Experts
"Thermohydraulic of Emergency Core Cooling in Light Water Reactors"
Report No. 161, Committee on the Safety of Nuclear Installations (Sep. 1989)

/2/ US-NRC
"Compendium of ECCS Research for Realistic LOCA Analysis"

"An overview of the code scaling, applicability, and uncertainty evaluation methodology"
J. Nuclear Engineering and Design Vol. 119, 1990

"Characterization of important contributors to uncertainty"
J. Nuclear Engineering and Design Vol. 119, 1990
"Evaluation of scale-up capabilities of best estimate codes"
J. Nuclear Engineering and Design Vol. 119, 1990

"A physically based method of estimating PWR large break loss of coolant accident PCT"
J. Nuclear Engineering and Design Vol. 119, 1990

7/ D'Auria F., Galassi G.M., Moschetti L.
"Assessment of Scaling Criteria adopted in Designing Nuclear Power Plants Experimental Simulators"
J. of Nuclear Materials, Vol. 130, 1985

8/ D'Auria F., Galassi G.M., Oriolo F., Vigni P.
"Assessment of Scaling Principles for the Simulation of Small Break LOCA Experiments in PWRs"

9/ D'Auria F., Karwat H., Mazzini M.
"Planning of Counterpart Tests in LWR Experimental Simulators"
25th National Heat Transfer Conf., Houston (TX), July 24-27 1988

10/ Bovalini R., D'Auria F.
"Scaling of the accuracy of Relap5/mod2 Code"
J. Nuclear Engineering and Design, vol 139 Nr. 1, 1993

11/ Bovalini R., D'Auria F., Galassi G.M.
"Scaling of complex phenomena in System Thermalhydraulics"

12/ D'Auria F.
"Experimental Facilities and System Codes in Nuclear Reactor Safety"
Int. Seminar State of the Art on Safety Analyses and Licensing of Nuclear Power Plants, Varna (BG), Nov. 2-6 1987

13/ D'Auria F., Ferri R., Vigni P.
"Evaluation of the data base from the Small Break LOCA Counterpart tests performed in PWR Experimental Simulators"
11th Conf. of Italian Society of Heat Transport, Milan (I), June 24-26 1993

14/ D'Auria F., Galassi G.M
"Code Assessment Methodology and Results"
IAEA Technical Workshop / Committee on Computer Aided Safety Analyses - Moscow (USSR), May 14-17 1990

15/ Billi C., D'Auria F., Debrecin N., Galassi G.M.
"Application of Relap5/mod2 to PWR International Standard Problems"
ANS Winter Meeting, San Francisco (Ca), Nov. 10-15 1991

16/ D'Auria F.
"Scaling and counterpart tests"
University of Pisa Report, DCMN - NT 192(92), Pisa (I), March 1992
Presentation at OECD-CSNI Spec. Meet. On Transient Two-Phase Flow - System Thermalhydraulics - Aix-en-Provence (F), Apr. 6-8, 1992

17/ D'Auria F., Karwat H.
University of Pisa Report, DCMN - NT 138(89), Pisa (I), Aug. 1989
/18/ Ambrosini W., Bovalini R., D'Auria F.  
"Evaluation of Accuracy of Thermalhydraulic Codes Calculations"  
J. Energia Nucleare, Vol. 7 N. 2, May 1990

/19/ Leonardi M., D'Auria F., Pochard R.  
"Quantitative code accuracy evaluation of ISP 27 based on a BETHSY Experiment"  
2nd Code Assessment and Maintenance Program (CAMP) Meet., Bruxelles (B), May 10-13 1993

/20/ Bovalini R., D'Auria F., De Varti A., Mauger P., Mazzini M.  
"Analysis of Counterpart Tests performed in BWR Experimental Simulators"  
J. Nuclear Technology, Vol. 97 Nr. 1, 1992

/21/ D'Auria F., Galassi G.M., Vigni P., Calasti A.  
"Scaling of Natural Circulation in PWR Systems"  

/22/ D'Auria F., Debrecin N., Galassi G.M., Galeazzi S.  
"Application of RELAP5/MOD3 to the Evaluation of Loss of Feedwater in test Facilities and in Nuclear Plants"  

/23/ Annunziato A., Addabbo C., Briday G., Deruaz R., Juhel D., Kumamaru H., Kukita Y., Medich C., Rigamonti M.  
"Small break LOCA counterpart test in the LSTF, Bethsy, LOBI and SPES test facilities"  
5th Int. Topl. Mtg. Reactor Thermalhydraulics, Salt Lake City, Utah, September 21-24, 1992

/24/ D'Auria F., Ferri R., Galassi G.M., Sugaroni F.  
"Evaluation of the data base from the small break LOCA Counterpart tests performed in LOBI, SPES, BETHSY and LSTF Facilities"  
University of Pisa Report, DCMN - NT 193(92), Pisa (I), June 1992

/25/ D'Auria F., Di Marco P., Mazzini M., Vigni P.  
"Thermalhydraulic Design of PIPER-ONE Loop"  
ANS Winter Meet., San Francisco (CA), Nov. 14-18 1985

/26/ D'Auria F., Vigni P.  
"Proposed Set of Criteria in Designing Nuclear Power Plants Experimental Simulators"  
3rd Int. Top. Meet. on Reactor Thermalhydraulics, Newport (RI), Oct. 15-18 1985

/27/ Cioni L., D'Auria F., Di Marco P., Galassi G.M., Mazzini M.  
"PIPER-ONE research: boundary and initial conditions for OECD-CSNI International Standard Problem 21 (ISP 21)"  
OECD-CSNI 1st Workshop on ISP 21, Marina di Grosseto (I), Sept. 22-23 1986

/28/ D'Auria F., Galassi G.M.  
"Planning of Spe5 experiment SP-SB-03 counterpart to LSTF, Bethsy and Lobi- Mod2 experiments"  
University of Pisa Report, DCMN - NT 171(91), Pisa (I), May 1991

/29/ Bovalini R., D'Auria F., De Varti A., Mauger P., Mazzini M.  
"Analysis of counterpart tests performed in BWR experimental simulators"  
OECD-CSNI 2nd Workshop on ISP 21, Calci (I), Apr. 13-14, 1989

/30/ Bajs T., D'Auria F., Debrecin N., Ferri R., Galassi G.M.  
"Analysis of the RELAP5/MOD2 code calculations of KRKO "LSTF Kv scaled" and "Nominal" conditions small break LOCA in cold leg"  
University of Pisa Report, DCMN - NT 206(93), Pisa (I), Feb. 1993
/31/ Bonuccelli M., D'Auria F., Debrecin N., Galassi G.M.
"A methodology for the qualification of thermalhydraulic codes Nodalizations"
6th Int. Top. Meet. on Nuclear Reactor Thermalhydraulics, Grenoble (F), Oct. 5-8 1993

/32/ D'Auria F., Galassi G.M., Lombardi P.
"User and Models Interactions in System Codes Predictions"
3rd Int. Conf. on Simulation Methods in Nuclear Engineering, Montreal (C), Apr. 18-20 1990

/33/ Aksan S. N. D'Auria F., Staedtike H.
"User Effects on the Thermalhydraulic Transient System Codes Calculations"
CSNI Spec. Meet. On Transient Two-Phase Flow - System Thermalhydraulics, Aix-En-Provence (F), Apr. 6-8, 1992

/34/ D'Auria F., Debrecin N., Galassi G.M.
"Application of RELAP5/MOD2 system code to small break LOCA counterpart experiments in PWR simulators"
IX Brazilian Meet. on Reactor Physics and Thermalhydraulics, Caxambu, Minas Gerais (BRA), Oct. 25-29 1993

/35/ D'Auria F., Calastri A., Galassi G.M.
"Scaling Capabilities of Thermalhydraulic System Codes"
Int. Conf. on Nuclear Engineering (ICONE-1) - Tokyo (J), Nov. 4-7 1991

/36/ D'Auria F.
"Effects of code installations on the predicted results"

/37/ Crommelynck Y.
"Computer effects on code results, code Relap5/Mod2/V251"
Presentation at the 9th OECD-CSNI THSB Task Group Meeting, Paris, December 16-18, 1992

/38/ D'Auria F.
"Conceptual Design of a PWR Experimental Simulator"
Int. Conf. on Safety and Advancements of Nuclear Power Plants, Varna (BG), Oct. 6-10 1986

/39/ Fletcher C.D., Schultz R.R.
"Relap5/Mod3 code Manual, Volume V: user's guide lines"
NUREG/CR-5535, August 1991

/40/ D'Auria F., Debrecin N., Galassi G.M., Lombardi P.
"Qualification process of Bethsy nodialization for Relap5/mod2 Code (Steady- State Level)"
University of Pisa Report, DCMN - NT 169(91), Pisa (I), March 1991

/41/ D'Auria F., Vigni P.
"Proposed Set of Criteria in Designing Nuclear Power Plants Experimental Simulators"
3rd Int. Top. Meet. on Reactor Thermalhydraulics, Newport (RI), Oct. 15-18 1985

/42/ CSNI Group of Experts
"CSNI Code Validation Matrix of Thermo-Hydraulic Codes for LWR LOCA and Transients"
CSNI Report N. 132, Paris (F), March 1987

/43/ Belsite S., D'Auria F., Galassi G.M.
"Evaluation of the data base from computer codes calculations of small break LOCA counterpart tests performed in LOBI, SPES, BETHSY and LSTF facilities"
University of Pisa Report, DCMN - NT 205(93), Pisa (I), May 1993

/44/ D'Auria F., Leonardi M.
"Quantitative accuracy evaluation of counterpart test data base"
University of Pisa, DCMN - NT 213(93), Pisa (I), September 1993
D'Auria F.
"Outline of UMAE: (Uncertainty Methodology based on Accuracy Extrapolation)"
Presentation at the 10th Meet. of CSNI Task Group on Thermalhydraulic System Behaviour,
Paris (F), June 30-July 2, 1993

Belsito S., D'Auria F., Galassi G. M.
"Application of a Statistical model to the evaluation of Counterpart Test data base"
ANS Winter Meeting, San Francisco (US), Nov. 10-13 1993

Bajs T., Debrecin N., D'Auria F., Galassi G. M.
"Uncertainty evaluation for RELAP5/MOD2 code calculations of the NPP KRSKO small break
LOCA scenario"
University of Pisa Report, DCMN - NT 209(93), Pisa (I), June 1993

Holmstrom H.
"Quantification of code uncertainty"
OECD-CSNI Spec. Meet. On Transient Two-Phase Flow - System Thermalhydraulics - Aix-en-
Provence (F), Apr. 6-8, 1992

Ghan L. S., Ortiz M.G.
"Uncertainty analysis of minimum vessel liquid inventory during a small-break LOCA in a
B&W plant - An application of the CSAU methodology using the Relap5/Mod3 computer code"
NUREG/CR-5818 EGG-2665, December 1992

Mavko B., Prosek A., Stritar A.
"Application of code scaling applicability and uncertainty methodology to large break LOCA
analysis of two loop PWR"
J. Nuclear Engineering and Design - to be issued

D'Auria F., Galassi G. M., Bajs T., Cavlina N., Debrecin N.
"Scaling of the Accident management Procedures during a Loss of Feedwater"
(CA), Nov. 8-13, 1992

Mayinger F., Weiss P., Wolfert K.
"Two-Phase Flow Phenomena in Full-Scale Reactor Geometry"
OECD-CSNI Spec. Meet. On Transient Two-Phase Flow - System Thermalhydraulics - Aix-en-
Provence (F), Apr. 6-8, 1992

Glaeser H., Karwat H.
"The Contribution of UPTF Experiments to resolve some scale-up Uncertainties in Countercurrent
Two-Phase Flow"
OECD-CSNI Spec. Meet. On Transient Two-Phase Flow - System Thermalhydraulics - Aix-en-
Provence (F), Apr. 6-8, 1992

Akimoto H., Iguchi T., Iwamura T., Murao Y.
"Large-Scale Multi-Dimensional Phenomena found in CCTF and SCTF Experiments"
OECD-CSNI Spec. Meet. On Transient Two-Phase Flow - System Thermalhydraulics - Aix-en-
Provence (F), Apr. 6-8, 1992

D'Auria F., Galassi G.M.
"Characterization of Instabilities during Two-Phase Natural Circulation in PWR Typical
Conditions"
J. Experimental Thermal and Fluid Science, Vol 3, Nr 90, 1990
<table>
<thead>
<tr>
<th>Event</th>
<th>Unit</th>
<th>LOBI</th>
<th>SPES</th>
<th>BETHSY</th>
<th>LSTF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Break</td>
<td>s</td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Scram</td>
<td>s</td>
<td>0.5</td>
<td>6.5</td>
<td>8</td>
<td>6</td>
</tr>
<tr>
<td>Pump trip</td>
<td>s</td>
<td>1.1</td>
<td>0</td>
<td>8</td>
<td>0</td>
</tr>
<tr>
<td>Feed water closure</td>
<td>s</td>
<td>12.7</td>
<td>15.5</td>
<td>8</td>
<td>13</td>
</tr>
<tr>
<td>Steam line closure</td>
<td>s</td>
<td>-</td>
<td>6.5</td>
<td>8</td>
<td>11</td>
</tr>
<tr>
<td>Accumulator intervention</td>
<td>s</td>
<td>420</td>
<td>355.5</td>
<td>345</td>
<td>346</td>
</tr>
<tr>
<td>LPIS intervention</td>
<td>s</td>
<td>2100</td>
<td>1522.5</td>
<td>not actuated</td>
<td>not actuated</td>
</tr>
<tr>
<td>End of the transient</td>
<td>s</td>
<td>2400</td>
<td>2034</td>
<td>2179</td>
<td>2113</td>
</tr>
</tbody>
</table>

*LPIS switched on following high heater rod temperature signal

**TAB. 1 - Boundary conditions during the counterpart tests**
<table>
<thead>
<tr>
<th>TSE/SVP</th>
<th>unit</th>
<th>LOBI</th>
<th>SPES</th>
<th>BETHSY</th>
<th>LSTF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Duration of phase a)</td>
<td>s</td>
<td>0±164</td>
<td>0±138</td>
<td>0±176</td>
<td>0±165</td>
</tr>
<tr>
<td>time of pressurizer emptying</td>
<td>s</td>
<td>0-23</td>
<td>0-12</td>
<td>0-18</td>
<td>0±16</td>
</tr>
<tr>
<td>maximum break flow rate/initiation</td>
<td></td>
<td>4.72</td>
<td>2.71</td>
<td>2.38</td>
<td>1.75</td>
</tr>
<tr>
<td>average specific break flow rate</td>
<td>kg/sm²</td>
<td>39706</td>
<td>50257</td>
<td>47872</td>
<td>48169</td>
</tr>
<tr>
<td>first dry-out duration</td>
<td>s</td>
<td>135-178</td>
<td>112-132</td>
<td>134-169</td>
<td>94-129</td>
</tr>
<tr>
<td>period for dry-out starting at top</td>
<td>s</td>
<td>no dry-out</td>
<td>no dry-out</td>
<td>no dry-out</td>
<td>no dry-out</td>
</tr>
<tr>
<td>level</td>
<td></td>
<td>154-157</td>
<td>112-134</td>
<td>124-129</td>
<td>97-105</td>
</tr>
<tr>
<td>period for dry-out starting at top</td>
<td>s</td>
<td>135-167</td>
<td>106-134</td>
<td>117-142</td>
<td>94-102</td>
</tr>
<tr>
<td>level</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>time of pump stop or pump velocity</td>
<td>s (rpm)</td>
<td>11.5 (0)</td>
<td>7 (0)</td>
<td>(2)</td>
<td>0 (0)</td>
</tr>
<tr>
<td>direction of bypass flow rate</td>
<td></td>
<td>from UH to DC</td>
<td>from UH to DC</td>
<td>from UH to DC</td>
<td>from UH to DC</td>
</tr>
<tr>
<td>break two phase flow starts</td>
<td>s</td>
<td>70±170</td>
<td>80-140</td>
<td>130-200</td>
<td>130-210</td>
</tr>
<tr>
<td>loops where clearing occurs</td>
<td></td>
<td>all loops</td>
<td>loop2</td>
<td>all loops</td>
<td>all loops</td>
</tr>
<tr>
<td>pressure drop in LSDL &lt; 10 KPa</td>
<td>s</td>
<td>112±140</td>
<td>102±107</td>
<td>116±123</td>
<td>101</td>
</tr>
<tr>
<td>occurrence of natural circulation</td>
<td></td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>cold leg flow rate is less than 5%</td>
<td>s</td>
<td>157</td>
<td>110</td>
<td>106</td>
<td>180</td>
</tr>
<tr>
<td>of the initial value</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Duration of phase b)</td>
<td>s</td>
<td>164±1706</td>
<td>138±1322</td>
<td>176±1730</td>
<td>165±1572</td>
</tr>
<tr>
<td>time of primary-secondary pressure</td>
<td>s</td>
<td>164</td>
<td>115-138</td>
<td>176</td>
<td>165</td>
</tr>
<tr>
<td>reversal</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>time of SGs U-tubes emptied</td>
<td>s</td>
<td>450</td>
<td>150</td>
<td>200</td>
<td>250</td>
</tr>
<tr>
<td>direction of bypass flow rate</td>
<td></td>
<td>from UH to DC</td>
<td>from UH to DC</td>
<td>from UH to DC</td>
<td>from UH to DC</td>
</tr>
<tr>
<td>second dry-out duration</td>
<td>s</td>
<td>362-550</td>
<td></td>
<td>342-367</td>
<td></td>
</tr>
<tr>
<td>period for dry-out starting at top</td>
<td>s</td>
<td>no dry-out</td>
<td>no dry-out</td>
<td>no dry-out</td>
<td>no dry-out</td>
</tr>
<tr>
<td>level</td>
<td></td>
<td>362-445</td>
<td>no dry-out</td>
<td>342-361</td>
<td>no dry-out</td>
</tr>
<tr>
<td>period for dry-out starting at high</td>
<td>s</td>
<td>850-1070</td>
<td>550-980</td>
<td>400-1200</td>
<td>400-1150</td>
</tr>
<tr>
<td>level</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>primary mass at time of accumulator</td>
<td>%</td>
<td>21</td>
<td>26</td>
<td>21.4</td>
<td>19.8</td>
</tr>
<tr>
<td>intervention</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>accumulator intervention period</td>
<td>s</td>
<td>420-953</td>
<td>355-309</td>
<td>345-976</td>
<td>346-688</td>
</tr>
<tr>
<td>average specific break flow rate</td>
<td>kg/sm²</td>
<td>3606</td>
<td>3568</td>
<td>2840</td>
<td>2605</td>
</tr>
<tr>
<td>saturation temperature decrease in</td>
<td>K</td>
<td>559-548</td>
<td>560-549</td>
<td>560-554 (+)</td>
<td>560-557</td>
</tr>
<tr>
<td>SG-SS</td>
<td></td>
<td>561-532</td>
<td>561-532</td>
<td>561-552 (+)</td>
<td>561-552 (+)</td>
</tr>
<tr>
<td>time when minimum mass in primary</td>
<td>s</td>
<td>420</td>
<td>355</td>
<td>350</td>
<td>350</td>
</tr>
<tr>
<td>side occurs</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Duration of phase c)</td>
<td>s</td>
<td>1705±2100</td>
<td>1322±1522</td>
<td>1730±2179</td>
<td>1572±2113</td>
</tr>
<tr>
<td>third dry-out duration</td>
<td>s</td>
<td>1705-2150</td>
<td>1322-1500</td>
<td>1730-end</td>
<td>1572-end</td>
</tr>
<tr>
<td>period for dry-out starting at top</td>
<td>s</td>
<td>no dry-out</td>
<td>no dry-out</td>
<td>no dry-out</td>
<td>no dry-out</td>
</tr>
<tr>
<td>level</td>
<td></td>
<td>1907-1914</td>
<td>1444-1446</td>
<td>2098-2149</td>
<td>1870-1884</td>
</tr>
<tr>
<td>period for dry-out starting at high</td>
<td>s</td>
<td>1673-1959</td>
<td>1322-1401</td>
<td>1730-1819</td>
<td>1572-1634</td>
</tr>
<tr>
<td>level</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>duration of phase d)</td>
<td>s</td>
<td>2100±2400</td>
<td>1522±2034</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>time of minimum mass in the primary</td>
<td>s</td>
<td>2100</td>
<td>1522</td>
<td>2100</td>
<td>2100</td>
</tr>
<tr>
<td>side</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>value of minimum mass in primary</td>
<td>%</td>
<td>21</td>
<td>26</td>
<td>21.4</td>
<td>19</td>
</tr>
<tr>
<td>side</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>saturation temperature decrease in</td>
<td>K</td>
<td>548-546</td>
<td>540-547</td>
<td>554-553</td>
<td>557-556</td>
</tr>
<tr>
<td>SG-SS</td>
<td></td>
<td>552-551</td>
<td>552-551</td>
<td></td>
<td></td>
</tr>
<tr>
<td>average specific break flow rate</td>
<td>kg/sm²</td>
<td>1764</td>
<td>2432</td>
<td>1006</td>
<td>1635</td>
</tr>
<tr>
<td>average rate of surface temperature</td>
<td>K/s</td>
<td>1.4</td>
<td>1.1</td>
<td>0.6</td>
<td>1.5</td>
</tr>
<tr>
<td>increase</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LPIS intervention</td>
<td></td>
<td>Yes</td>
<td>Yes</td>
<td>No</td>
<td>No</td>
</tr>
<tr>
<td>time of LPIS intervention</td>
<td>s</td>
<td>2100</td>
<td>1522</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>LPIS injected mass/facility volume</td>
<td>kg/m³</td>
<td>150</td>
<td>296</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>average specific break flow rate</td>
<td>kg/sm²</td>
<td>784</td>
<td>3839</td>
<td>1329</td>
<td>243</td>
</tr>
</tbody>
</table>

(+ in loops 1 and 3
(*) in loop 2

TAB. 2 - TSE and SVP values during the counterpart tests
<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>( U )</th>
<th>((U)_{95})</th>
</tr>
</thead>
<tbody>
<tr>
<td>time of pressurizer emptying</td>
<td>( \pm 3.3 , \text{s} )</td>
<td>( \pm 9 , \text{s} )</td>
</tr>
<tr>
<td>time of first dry-out</td>
<td>( \pm 29 , \text{s} )</td>
<td>( \pm 53 , \text{s} )</td>
</tr>
<tr>
<td>time of secondary side-to-primary side pressure reversal</td>
<td>( \pm 10 , \text{s} )</td>
<td>( \pm 23 , \text{s} )</td>
</tr>
<tr>
<td>time of loop seal clearing</td>
<td>( \pm 33 , \text{s} )</td>
<td>( \pm 107 , \text{s} )</td>
</tr>
<tr>
<td>average specific break flowrate during phase a)</td>
<td>( \pm 4701 , \text{kg/s m}^2 )</td>
<td>( \pm 11223 , \text{kg/s m}^2 )</td>
</tr>
<tr>
<td>minimum primary mass over facility volume</td>
<td>( \pm 41 , \text{kg/m}^3 )</td>
<td>( \pm 110 , \text{kg/m}^3 )</td>
</tr>
<tr>
<td>time of third dry-out</td>
<td>( \pm 330 , \text{s} )</td>
<td>( \pm 700 , \text{s} )</td>
</tr>
<tr>
<td>time when pressurizer pressure reaches 4 MPa</td>
<td>( \pm 26 , \text{s} )</td>
<td>( \pm 64 , \text{s} )</td>
</tr>
<tr>
<td>primary pressure at time of start of third dry-out</td>
<td>( \pm 0.189 , \text{MPa} )</td>
<td>( \pm 0.453 , \text{MPa} )</td>
</tr>
<tr>
<td>modified Stanton number</td>
<td>( \pm 0.1 )</td>
<td>( \pm 0.23 )</td>
</tr>
<tr>
<td>Richardson number</td>
<td>( \pm 15 )</td>
<td>( \pm 51 )</td>
</tr>
<tr>
<td>Biot number</td>
<td>( \pm 0.14 )</td>
<td>( \pm 0.79 )</td>
</tr>
<tr>
<td>heat source number</td>
<td>( \pm 2.83 \times 10^{-4} )</td>
<td>( \pm 9.38 \times 10^{-4} )</td>
</tr>
<tr>
<td>integral of primary pressure</td>
<td>( \pm 483 , \text{MPa s} )</td>
<td>( \pm 1241 , \text{MPa s} )</td>
</tr>
<tr>
<td>integral of primary mass over facility volume</td>
<td>( \pm 82072 , \text{kgs/m}^3 )</td>
<td>( \pm 185334 , \text{kgs/m}^3 )</td>
</tr>
</tbody>
</table>

**TAB. 3** - 'Average uncertainty' and '95th percentile' uncertainties obtained in relation to selected TSE, SVP and IPA
<table>
<thead>
<tr>
<th>ITEM</th>
<th>GENERAL APPLICABILITY</th>
<th>APPLICABILITY TO THE PRESENT CASE</th>
<th>NOTES</th>
</tr>
</thead>
<tbody>
<tr>
<td>User effects (*)</td>
<td>To be considered</td>
<td>No</td>
<td>Modelling of relief valves and of components and systems not used in this study</td>
</tr>
<tr>
<td>Nodalization details (+)</td>
<td></td>
<td></td>
<td>Core, downcomer, steam generator tubes, etc.</td>
</tr>
<tr>
<td>Single channel</td>
<td>No</td>
<td>No</td>
<td></td>
</tr>
<tr>
<td>Multiple parallel channels</td>
<td>To be considered</td>
<td>Not necessary</td>
<td></td>
</tr>
<tr>
<td>Occurrence of 2D/3D effects</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core</td>
<td></td>
<td>To be considered</td>
<td></td>
</tr>
<tr>
<td>Core/Upper plenum</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Downcomer</td>
<td></td>
<td>To be considered</td>
<td></td>
</tr>
<tr>
<td>Hot leg and cold leg piping</td>
<td></td>
<td>No</td>
<td></td>
</tr>
<tr>
<td>Steam generator tubes</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Modelling of nuclear fuel</td>
<td>To be considered</td>
<td>To be considered</td>
<td>It gives negligible contribution in the present case</td>
</tr>
</tbody>
</table>

(*) Uncertainties in boundary and initial conditions are part of these items
(+) The nodalization qualification procedure must be followed

TAB. 4 - Additional biases to be considered in the UMAE
FIG. 1 - Possible trends resulting from scaling analyses:
extrapolation of measured events/data and of code accuracy
FIG. 2 - Exploitation of counterpart test data
General qualification process

Plant nodalization

Code

GI

FG

ITF nodalizations

Specific experimental data

ITF calculations

Accuracy quantification

Accuracy extrapolation

Demonstration of similarity

(Phenomena analysis)

(Scaling laws)

Yes

ASM calculation

Uncertainty

(*) Special methodology developed

(*) Dashed blocks represent blocks that are common to UMAE and CSAU.

FIG. 3 - Simplified flowsheet of UMAE
General qualification process → a Code → Code applicability

Engineering judgement (e.g. number of nodes) → d Plant nodalization

Engineering judgement (e.g. selection of a given parameter) → e

Selection of the input parameters → f

Phenomena analysis
Scaling laws
Accuracy

Plant calculation → h

Sensitivity analysis → i

Response surface → j

Uncertainty

(*) Dashed blocks represent blocks that are common to UMAE and CSAU.

FIG. 4: Simplified flowsheet of CSAU
FIG. 5 - Small break LOCA counterpart tests: pressurizer pressure

FIG. 6 - Small break LOCA counterpart tests: scaled integral break flowrate

FIG. 7 - Small break LOCA counterpart tests: heater rod surface temperature

FIG. 8 - Small break LOCA counterpart tests: scaled primary side residual mass
FIG. 9 - Scaling of counterpart tests parameters: peak cladding temperature at time of first dry out

FIG. 10 - Scaling of counterpart tests parameters: time of loop seal clearing

FIG. 11 - Scaling of counterpart tests parameters: time of accumulators intervention

FIG. 12 - Scaling of counterpart tests parameters: time of third dry-out
FIG. 13 - Rod surface temperature at peak axial flux location: reference ASM calculation result related to KRSKO reactor and envelope of 'time', 'spatial' and 'mean integral' uncertainty values.

FIG. 14 - Integral of break flowrate: reference ASM calculation result related to KRSKO reactor and envelope of 'time', 'spatial' and 'mean integral' uncertainty values.
UNIVERSITY OF PISA

DIPARTIMENTO DI COSTRUZIONI
MECCANICHE E NUCLEARI

THE FFT BASED METHOD UTILISATION
IN THE FRAME OF THE UMAE

M. Leonardi (+), F. D'Auria (+), R. Pochard (#)

(+) University of Pisa - DCMN - via Diotisalvi 2 - 56100 Pisa - Italy
(#) CEA IPSN DPEI SEAC - BP 6, 92265 Fontenay aux Roses - France

Presented at the "Special Workshop on Uncertainty Analysis Methods"
THE FFT BASED METHOD UTILISATION IN THE FRAME OF THE UMAE

M. Leonardi (†), F. D'Auria (†), R. Pochard (§)

(†) University of Pisa - DCMN - via Diotisalvi 2 - 56100 Pisa - Italy
(§) CEA IPSN DPEI SEAC - BP 6, 92265 Fontenay aux Roses - France

ABSTRACT

The purpose of this paper is to outline the use of the FFT (Fast Fourier Transform) based method in the UMAE (Uncertainty Methodology based on Accuracy Extrapolation), methodology suitable to evaluate the uncertainty in the prediction, by thermalhydraulic system codes, of the transient scenarios expected in nuclear reactors.

The UMAE methodology is based on the extrapolation of the accuracy resulting from the comparison between code results and relevant data gathered in scaled facilities.

The evaluation of quantitative code accuracy is obtained by a methodology utilising the FFT in order to relate the discrepancies between experimental and code calculated trends in the frequency domain.

A global outline of the FFT method is here presented, along with the field of application in the UMAE context. A few relevant results concerning the performed and in progress applications are also reported.

1. INTRODUCTION

The quantitative evaluation of code uncertainties is a necessary step in the code assessment process, above all if BE (Best Estimate) codes are utilised for licensing purposes. In fact, if BE codes allow safety margins reductions, taking into account more realistic conditions, on the other hand, their utilisation requires the evaluation of all the uncertainties related with performed calculations.

The large number of phenomena occurring in a NPP (Nuclear Power Plant) during a thermalhydraulic transient, needs of very complex calculation models, in order to represent the whole plant apparatus behaviour. As a consequence, the reliability of these code predictions, adopted for safety analyses of NPPs, depends on many factors involving code features and user experience.

The OECD/CSNI International Standard Problem (ISP) program was settled to investigate the validity and the accuracy (i.e. the capability to correctly predict the observed scenario of tests performed in scaled facilities) of used computer codes and to increase the general confidence in their utilisation. But, the evaluation of the prediction capabilities of a code requires, due to the limits in the present knowledge of the scaling laws and to the lack of suitable measurements from the plants, the definition of the uncertainty related to the unknown error performed in predicting plant behaviour. /1/.
The assessment process of thermalhydraulic system codes aims mainly at verifying the goodness of code predictions against experimental data gathered principally by tests performed in separate effects and integral plant simulators, /2/.

In order to evaluate the applicability of a code in predicting a plant scenario, one should be sure that, at least, the experimental data used for qualifying the codes are representative of phenomena expected in the plant; then, that codes are able to reproduce qualitatively and quantitatively these data.

Starting from these basic considerations, the UMAE methodology has been developed. The UMAE is based upon the extrapolation of the accuracy evaluated by the comparison between code results and experimental data from scaled facilities, to get the uncertainty in the plant scenario.

The quantification of code accuracy (in the UMAE) is pursued through the utilisation of a special approach, developed at the University of Pisa (DCMN). It is an integral method using the FFT in order to represent the code discrepancies in the frequency domain.

A global outline of the FFT method is here presented, along with the field of applicability and use in the UMAE context. A few relevant results concerning the performed and in progress applications are also reported.

2. FFT METHOD BASICS

In the aim to identify one or more indexes quantifying the accuracy, several methods have been proposed. The simplest formulation about the accuracy of a given calculation is obtained by the difference function:

\[ \Delta F(t) = F_{\text{calc}}(t) - F_{\text{exp}}(t). \]  

(1)

The information provided by this time dependent function, continuously varying, should be condensed to give a limited number of values that could be taken as indexes for quantifying accuracy. Integral approaches satisfy this requirement, since they produce a single value on the basis of the instantaneous trend of a given time function. On the other hand, careful should be involved in expressing all the information through a single value, in such a way to avoid the loss of significant details, /3/.

A fundamental property of the Fourier Transform is that we can analyse a particular relationship from a completely different viewpoint, without any lack of information with respect to the original one. Moreover, the time trend of a given parameter can hid the presence of perturbations, otherwise easily visible through an harmonic analysis.

When using functions sampled in digital form, the FFT can be used, i.e. an algorithm that computes more rapidly the discrete Fourier Transform. To apply this algorithm, functions must be identified by a number of values which is a power of 2. Thus, if the number of points defining the function in the time domain is \( N=2^{m+1} \), the FFT gives the transformed function defined in the frequency domain by \( 2^{m+1} \) values associated to the frequencies \( f_n=n/T, \ (n=0,1,...,2^m) \), in which \( T \) is the time duration of the sampled signal. Supposing that available data are characterised by an adequate sampling frequency, the fulfilment of the Sampling Theorem is required to avoid distortion of sampled signals, /4/.

If we select \( N=2^{m+1} \) points, the maximum frequency of FFT transformed functions is given by:
\[ f_{\text{max}} = \frac{2^m}{T_d} = \frac{f_s}{2} \]  

where
- \( T_d \) is the transient time duration,
- \( f_s \) is the sampling frequency.

In other words, it is meaningless to choose a number of points giving, in terms of FFT analysis, a frequency greater than the maximum one achievable adopting a certain \( f_s \). On the other hand, some information could be lost using a too low number of points.

The accuracy quantification of a code calculation is based on the amplitude of the FFT of the experimental signal and of the difference between this one and the calculated trend. In particular, the method characterises each calculation through two values:

- **a dimensionless Average Amplitude (AA)**

\[
\text{AA} = \frac{\sum_{n=0}^{2^m} |\Delta F(f_n)|}{\sum_{n=0}^{2^m} |F_{\exp}(f_n)|}
\]  

- **a Weighted Frequency (WF)**

\[
\text{WF} = \frac{\sum_{n=0}^{2^m} |\Delta F(f_n)| \cdot f_n}{\sum_{n=0}^{2^m} |\Delta F(f_n)|}
\]

The most significant information is given by AA, which represents the relative magnitude of the discrepancy deriving from the comparison between the addressed calculation and the corresponding experimental trend (AA=1 means a calculation affected by a 100% of error). The WF factor characterises the kind of error, because its value emphasises if the error has more relevance at low or high frequencies, and depending on transient, high frequency errors can be more acceptable than low frequency ones (in other words, analysing thermal hydraulic transients, better accuracy is generally represented by low AA values at high WF values).

A cut frequency has been introduced, characterising the upper frequency value which has to be considered in evaluating the AA and WF factors, as defined by eqs. (3) and (4). It has been considered to filter spurious contributions, generally negligible (e.g. slopes added to interpolated signals, as an effect of the performed linear interpolation, necessary to have a power of 2 number of points). This value is obviously related to the frequencies characterising the addressed parameters and the available data. Analyses performed, /5/, on typical thermal hydraulic quantities show that typical values for this quantity are in the range 0.3 - 0.5 Hz.
Trying to give an overall picture of the accuracy of a given calculation, average indexes of performance are obtained by defining:

\[
(AA)_{\text{tot}} = \sum_{i=1}^{N_{\text{var}}} (AA)_i \cdot (w_p)_i
\]  
(5)

\[
(WF)_{\text{tot}} = \sum_{i=1}^{N_{\text{var}}} (WF)_i \cdot (w_p)_i
\]  
(6)

where \(N_{\text{var}}\) is the number of analysed parameters and \((w_p)_i\) are weighting factors introduced to take into account the different importance of each parameter from the viewpoint of safety analyses. Briefly, each \((w_p)_i\) takes into account:

- "experimental accuracy": experimental trends of thermal hydraulic parameters are characterised by a more or less sensible uncertainty due to:
  
  + intrinsic characteristics of measure instruments
  + determination method of the measure
  + different evaluation ways between experimental measures and the code calculated ones;

- "safety relevance": particular importance is given to the accuracy evaluation of code calculations concerned with those parameters (such as pressure, peak clad temperature, etc.) which are relevant for safety and design.

Further contribution is given by a factor which normalises the AA value calculated for the selected parameters with respect to the AA value calculated for primary side pressure. This factor has been introduced in order to consider the intrinsic and physic relations existing between different quantities (i.e. fluid temperatures and pressures must be characterised by the same order of error). So doing, the weighting factor of the j-th parameter is defined as:

\[
(w_p)_j = \frac{(w_{\text{exp}})_j \cdot (w_{\text{saf}})_j \cdot (w_{\text{norm}})_j}{\sum_{j=1}^{N_{\text{var}}} (w_{\text{exp}})_j \cdot (w_{\text{saf}})_j \cdot (w_{\text{norm}})_j}
\]  
(8)

where:
- \(w_{\text{exp}}\) is the contribution related to the experimental accuracy;
- \(w_{\text{saf}}\) is the contribution which expresses the safety relevance of the addressed parameter;
- \(w_{\text{norm}}\) is the component of normalisation with reference to the average amplitude evaluated for the primary side pressure.
This introduces some degree of engineering judgement that can be partly reduced by a proper and unique definition of the weighting factors. Then, a set of weights has been defined for typical thermalhydraulic quantities; it is summarised in Tab. 1. These values are the result of many considerations involving among the other things problems related with different ways of obtaining experimental measures and corresponding code calculated quantities.

3. CODE ACCURACY EVALUATION

Evaluating the code accuracy commonly requires the accomplishment of the following two steps:
- qualitative level determination;
- quantitative level determination.

Concerning the qualitative level, the DCMN has set up a procedure for this kind of analysis, that is similar to the one suggested by CSNI, /6/, or by US INEL, /7/, including:

1) subdivision in "phenomenological windows";
2) for each phenomenological window:
   - specification of key phenomena;
   - identification of the "Relevant Thermalhydraulic Aspects" (RTAs);
   - selection of the parameters characterising the RTAs;
3) qualitative analysis of obtained results by comparing (only by a visual observation) experimental and calculated trends.

The qualitative analysis is also based on four subjective judgement marks, /7/:
- excellent (the code predicts qualitatively and quantitatively the parameter);
- reasonable (the code predicts only qualitatively the parameter);
- minimal (the code does not predict the parameter, but the reasons are understood and predictable);
- unqualified (the code does not predict the parameter, reasons are unknown);

through which we can operate a first classification about the calculation quality. This analysis is a necessary prerequisite to the application of the quantitative analysis. In fact, it is a nonsense performing this last one, if the calculation is not qualitatively correct.

With reference to the quantitative level, the most suitable factor for the definition of an acceptability criterion is the average amplitude AA. With reference to the accuracy of a given calculation (see eq. (5)), we can define the following acceptability criterion:

\[(AA)_{\text{tot}} < K\] (9)

where K is an acceptability factor valid for the whole transient. As lower is the \((AA)_{\text{tot}}\) value, as better is the accuracy of the analysed calculation (i.e. the code prediction capability and acceptability is higher). On the other hand, \((AA)_{\text{tot}}\) should not exceed the unity in any part of the transient (AA=1 means a calculation affected by a 100% error).
Due to this requirement, the accuracy evaluation should be performed at different steps during the transient, to verify if this condition is not satisfied in any phase of it. With reference to experience gathered from previous applications of this methodology (roughly 200 code calculations analysed from about ten experiments), $K = 0.4$ has been chosen as reference threshold value identifying good accuracy of a code calculation. /8/. In fact, taking into account the previous applications, it can be observed that results in the range:

1) $(AA)_{tot} \leq 0.3$ characterise very good code predictions;
2) $0.3 < (AA)_{tot} \leq 0.5$ characterise good code calculations;
3) $0.5 < (AA)_{tot} \leq 0.7$ characterise poor code predictions;
4) $(AA)_{tot} > 0.7$ characterise very poor code calculations.

The evaluation of the code capability in the single parameter prediction can be performed by such a similar criterion; clearly, in this case the AA factor is the one calculated for the addressed parameter (see eq. (3)). Presently, some activity is devoted in order to fix upper acceptability limits to the AA values related to safety relevant parameters, like cladding temperature, primary pressure, primary residual mass.

4. USE OF THE FFT METHOD IN THE UMAE

The FFT method was initially developed in the aim of setting up a reliable method suitable for evaluating the accuracy. So, the initial utilisation domain of the method concerned the quantification of the accuracy of a code prediction compared with related experimental data. The parallel development of the other steps of the UMAE have successively suggested other potential application areas for the FFT methodology. The exploitation of these application areas allowed the completion of the frame of the links and the closure of inferential loops existing in the UMAE, in order to allow its full application and get the final evaluation of the uncertainty.

With reference to Tab. 2, /9/, reporting in a schematic way the basic steps of the UMAE, the following items were identified as potentially exploitable by the FFT method:

1. demonstration of "accuracy" of code calculation results (item 8);
2. demonstration of "similarity" of experimental data (item 7);
3. demonstration of "similarity" of experimental data and "facility scaled" plant calculation;
4. qualification of the nodalization and user.

Several applications of the FFT method have been performed, in order to allow the exploitation of the four items above listed, in such a way to investigate its applicability in the UMAE context. The related results and a list of these applications are reported in the following.
4.1 Demonstration of "accuracy" of code calculation results

The previous applications of the FFT method concerned some ISPs related to SBLOCA situations both in PWR and in BWR, besides a LOFW transient in PWR.
In particular:

- the method has been partially applied to ISP 18 submitted code calculations. The selected test was the A2-81 performed in LOBI/MOD2 facility; a SBLOCA originated by a 1% \( A_{\text{max}} \) break in cold leg. RTAs was identified and the main parameters were analysed by the FFT method, /10/;

- the method has been applied to calculations submitted to ISP 21, a typical SBLOCA sequence in BWR; this experiment (test PO-SB-7) was performed in PIPER-ONE facility, and was characterised by a 2.6% \( A_{\text{sec}} \) break (in a recirculation line). The analysis of the main parameters (as primary pressure trend, heater rod temperatures, primary residual mass, core collapsed level, etc.) was performed on double blind submitted calculations and on post-test calculation performed by the host organisation, /11/;

- the method has been applied to analyse the ISP 22 submitted calculations. The test was a LOFW sequence (in all the 3 SGs, with delayed EFW actuation), and was performed in SPES facility (test SP-FW-02). The accuracy was evaluated over a set of 14 selected variables in double blind and post-test calculations, /12/;

- the method has been fully applied to ISP 27 blind and post-test calculations. The selected experiment was the test 9.1.b performed in BETHSY facility, and was a SBLOCA originated by 2" break in cold leg (loop with pressurizer) with an ultimate recovery procedure (steam dumps opening when heater rods temperature reached 773 K). More than 40 code calculations were analysed, the global code accuracy was evaluated on 25 selected parameters. The sensitivity of the methodology was investigated, the influence of the number of parameters to be analysed was demonstrated, /2/;

- the methodology has been fully applied to a counterpart test performed in SPES (test sp-sb-03), LOBI/MOD2 (test bl-34), BETHSY (test 6.2.tc), and LSTF (test sb-cl-21) facilities. The test is a typical SBLOCA sequence, originated by a 2" rupture in cold leg (loop with pressurizer). All the calculations were performed by the same qualified user, adopting qualified nodalizations; the adopted code was the RELAPS/MOD2. A sketch of the results obtained for some relevant parameters among the 24 analysed by the FFT method is reported in Tab. 3. The application of the method highlighted that the code is adequate to predict the transient scenario, and obtained results improve increasing the dimension of the facilities, /8/. A similar activity is in progress to evaluate the code accuracy concerning the CATHARE 2 V1.3 code calculations of this counterpart test.

Among the objectives related to the application of the FFT method in quantifying the code accuracy of thermalhydraulic code calculations, it is worth to mention that some activities are in progress to qualify the method applicability to a wider spectrum of transients. In particular, the following applications are in progress:
1. analysis of ISP 33 submitted code calculations (natural circulation test in WWERs, performed in the PACTEL facility);

2. analysis of ISP 35 submitted code calculations (NUPEC); the relevance of this application is related to the interest in investigating the sensitivity of the method for containment phenomenology's, typically slower than primary circuit ones;

3. analysis of a LOFW transient (LSTF experiment).

4.2 Demonstration of "similarity" of experimental data

This is one of the basic requirements to allow the application of the UMAE methodology, because it is meaningless, in counterpart test activities, the comparison of data which do not present the same phenomena. With reference to the time dependent parameters characterising the identified RTAs, the FFT method can be utilised to demonstrate the similarity of available experimental data. The method has been applied in order to verify if the counterpart experimental data satisfied this requirement. The LSTF experimental data were considered as the reference "experimental" data base. The application of the FFT method confirmed the similarity among these sets of data. The application concerned the 24 selected parameters utilised to evaluate the accuracy of calculations performed by DCMN, /8/, with the RELAP5/MOD2 code (see sect. 4.1). Due to the different final phases that characterised the four experiments (only two of four ended with the intervention of the LPIS), and to the shorter duration of the SPES test, the analysis was limited to the first 1250 s. from the beginning of the transients, in order to let them to be homogeneously compared. An excerpt of main obtained results and the global AA values are reported in Tab. 4. It is worth to note that the analysis of these experimental data with the FFT method exhibits very low AA values, so that the "similarity" is also demonstrated from a quantitative point of view, through an objective tool.

4.3 Demonstration of "similarity" of experimental data and "facility scaled" plant calculation

Two plant calculations have to be carried out according to the UMAE methodology, /1/: the 'facility scaled' and the 'realistic conditions' calculation. The first one is characterised by boundary and initial conditions utilised as input, derived from those in the experimental facilities following the counterpart test scaling criteria. Nominal conditions should be utilised in the latter case. In the application of UMAE to the SBLOCA counterpart tests performed in LOBI, SPES, BETHSY and LSTF rigs, the KRSKO plant was used as reference in the attempt to extrapolate the data measured in the facilities to a plant, /13/, /14/. As in the previous situation, the method has been similarly applied to the selected quantities as predicted in the 'facility scaled' calculation with reference to the LSTF experimental data. Also in this case, the application of the method has allowed the demonstration of the "similarity", according to the low AA_{10T} value obtained by the 'LSTF K_{v} scaled' KRSKO calculation. The results of this application, related to the main analysed variables and the AA_{10T}, are synthesised in Tab. 4.
4.4 Qualification of the nodalization and user

The UMAE requires the use of qualified nodalizations, /9/, in the prediction of ITF and plant scenarios. The developed nodalizations must be qualified taking into account the comparison with hardware data, boundary and initial conditions, time trends of relevant quantities. The qualification of a nodalizations is achieved when, following a particular procedure including a double level of qualification, /15/, /16/, a 'steady state' and a 'transient level' of qualification are obtained.

The development and the qualification of a nodalization have to be performed by a qualified user. In relation to this last delicate item, relevant suggestions can be found in /17/, where a user qualification process is formulated. It is well known that the 'user effect' is a strong concern in the evaluation of code accuracy.

In this context, the FFT method constitutes a relevant checking tool to be applied in order to verify, quantitatively, if the qualification process (of the nodalization and / or the user) can be considered as achieved, or if changes (in the case of the nodalization) are needed to get better results. The relationships to be satisfied concern either the global acceptability criterion (5), or a particular parameter trend, the prediction of which is not correct, as remarkable by the related AA.

The nodalizations utilised in the application of the UMAE to the SBLOCA counterpart tests, have been previously qualified according to this double level procedure, /16/. The transients taken as reference for the four facilities were:

1. the LOBI test a2-81 (ISP-18), /10/;
2. the SPES test sp-fw-02 (ISP 22), /12/;
3. the BETHSY test 9.1.b (ISP 27), /18/;
4. the LSTF test sb-cl-18 (ISP 26), /19/.

The FFT method has been applied, in each case, in order to check the prediction capabilities obtained utilising these input decks, and verify if the condition on $A_{tot}$ is satisfied. The application concerned a ten of parameters; the main results are listed in Tab. 5. In any case a value of $A_{tot}$ greater than 0.4 was obtained.

4.5 Extrapolation of the accuracy

The application of the FFT method to counterpart test calculations, /8/, confirmed the feasibility of the accuracy extrapolation to get uncertainty.

The extrapolation of the accuracy for relevant time dependent quantities has been performed by means of a statistical approach, /8/, but the uncertainty ranges obtained are not still related to physical quantities, due to the nature of the AA factors.

As a consequence of this, some activities are in progress to close this process. In particular, the correlation of the Peak Cladding Temperature (PCT) to calculated AA and WF values seems to be effective in the solution of this problem.

With reference to rod cladding temperature trends typically foreseeable in LOCA conditions, different possible calculated trends have been postulated, each of these steeply varying in some phases during the transient. Since the main parameter relevant in safety analyses is the PCT value, the idea was to evaluate the relation existing between the AA
obtained by the FFT method in the analysis of a calculated cladding temperature and the related difference \[\text{PCT}_{\text{cal}} - \text{PCT}_{\text{exp}}\], in such a way to get possible the determination of the uncertainty in the PCT calculated value associated to the AA factor characterising the addressed code calculation. Then, starting from the AA factor (related to rod temperature trends, see eq. (3)), by means of the analysis of the envelop of all the possible situations that can be envisaged by the FFT method, the corresponding uncertainty in PCT predictions can be estimated. As an example of this, in fig. 1 a typical rod cladding temperature for an Intermediate Break LOCA is considered, together with six possible situations that can occur in the code prediction of this quantity. In particular:

1) the code overestimates or underestimates the real PCT value, but with accurate timing (case A);
2) the code predicts only in approximate way (just a sharp peak highlighting the heatup and rewetting phase) the phenomenon, with good timing (case B);
3) the code predicts with delay the dryout occurrence, overestimating or underestimating the PCT value (case C);
4) the code predicts the dryout early in the transient, overestimating or underestimating the PCT value (case D);
5) the code predicts the dryout phenomenon but with higher or lower initial and final temperatures (case E);
6) the code predicts the dryout occurrence with good timing, but overestimating or underestimating the PCT value, the initial and final temperatures (case F).

As already mentioned, the calculated trends have been considered varying in a discrete way the entity of the error in the prediction of the PCT (just the bounding curves are reported in fig. 1 for each case, including roughly ~40 sample curves). The results of the application of the FFT method to these sample cases are shown in fig. 2, in which we can observe the typical PCT uncertainty corresponding to an AA value.

At present, the obtained results are promising, and further refinements of this step are in progress, including the similar analysis of temperature trends concerning Small and Large Break LOCA situations.

5. CONCLUSIONS

A special methodology based on the use of the FFT has been developed at University of Pisa in order to quantify the accuracy of code calculations. It is an integral approach evaluating the discrepancies between experimental data and code calculated trends in the frequency domain, in such a way to allow a more detailed characterisation (through harmonic analysis) of addressed signals. This methodology has been applied to a roughly 200 code calculations from about ten experiments.

This method is an essential part of the UMAE, being a necessary tool to the closure of the inferential loops existing in the UMAE, in order to allow its full application and get the final evaluation of the uncertainty.

In particular, the FFT method utilisation in the context of the UMAE is related to the following basic items, characterising the UMAE application:

1. demonstration of "accuracy" of code calculation results;
2. demonstration of "similarity" of experimental data;

3. demonstration of "similarity" of experimental data and "facility scaled" plant calculation;

4. qualification of the nodalization and user.

The FFT method has been applied, in order to allow the exploitation of the four items above listed, in such a way to investigate its wider applicability in the frame of UMAE. The work performed and the related results have been presented in this report. A general conclusion that can be drawn is that the FFT method confirms its relevance in the field of the code accuracy quantification, and presents, for its intrinsic features, some others application areas that increase its relevance in the UMAE context.

REFERENCES


19/ D'Auria F., Galassi G.M., Billa C., Debrecin N.: OECD CSNI ISP 26 - post-test analysis of a Small Break LOCA on the LSTF facility performed by RELAP5/MOD2 code at University of Pisa in the frame of the collaboration with University of Zagreb and ENEA. University of Pisa Report, DCMN-NT 151(90), Pisa, January 1990.
TABLES AND FIGURES

<table>
<thead>
<tr>
<th></th>
<th>$W_{exp}$</th>
<th>$W_{saf}$</th>
<th>$W_{norm}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressure drops</td>
<td>0.7</td>
<td>0.7</td>
<td>0.5</td>
</tr>
<tr>
<td>Mass inventories</td>
<td>0.8</td>
<td>0.9</td>
<td>0.9</td>
</tr>
<tr>
<td>Flowrates</td>
<td>0.5</td>
<td>0.8</td>
<td>0.5</td>
</tr>
<tr>
<td>Primary pressure</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Secondary pressure</td>
<td>1.0</td>
<td>0.6</td>
<td>1.1</td>
</tr>
<tr>
<td>Fluid temperatures</td>
<td>0.8</td>
<td>0.8</td>
<td>2.4</td>
</tr>
<tr>
<td>Clad temperatures</td>
<td>0.9</td>
<td>1.0</td>
<td>1.2</td>
</tr>
<tr>
<td>Collapsed levels</td>
<td>0.8</td>
<td>0.9</td>
<td>0.6</td>
</tr>
<tr>
<td>Core power</td>
<td>0.8</td>
<td>0.8</td>
<td>0.5</td>
</tr>
</tbody>
</table>

$W_{exp}$: experimental accuracy  
$W_{saf}$: safety relevance  
$W_{norm}$: primary pressure normalization

Tab. 1: Weighting factor components for typical thermalhydraulic quantities, /5/.

<p>| | | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>0.</td>
<td>- Availability of frozen code and relevant experimental data</td>
<td></td>
</tr>
<tr>
<td></td>
<td>- Choice of reactor system and accident scenario</td>
<td></td>
</tr>
<tr>
<td>1.</td>
<td>- Demonstration of code and user capabilities</td>
<td></td>
</tr>
<tr>
<td>2.</td>
<td>- Demonstration of &quot;suitability&quot; of facilities design</td>
<td></td>
</tr>
<tr>
<td>3.</td>
<td>- Demonstration of &quot;suitability&quot; of test design (specific)</td>
<td></td>
</tr>
<tr>
<td>4.</td>
<td>- Demonstration of &quot;suitability&quot; of test data (specific)</td>
<td></td>
</tr>
<tr>
<td>5.</td>
<td>- Qualification of facilities and plant nodalization</td>
<td></td>
</tr>
<tr>
<td>6.</td>
<td>- Evaluation of experimental and calculated data base (specific)</td>
<td></td>
</tr>
<tr>
<td>7.</td>
<td>- Demonstration of &quot;similarity&quot; of experimental data</td>
<td></td>
</tr>
<tr>
<td>8.</td>
<td>- Demonstration of &quot;accuracy&quot; of code calculation results</td>
<td></td>
</tr>
<tr>
<td>9.</td>
<td>- Execution of plant calculation (&quot;facility scaled&quot;) and demonstration of &quot;similarity&quot; with respect to facilities related results</td>
<td></td>
</tr>
<tr>
<td>10.</td>
<td>- Execution of plant calculation (&quot;realistic conditions&quot;) with the Analytical Simulation Model and demonstration that phenomena are the same as at item 9</td>
<td></td>
</tr>
<tr>
<td>11.</td>
<td>- Uncertainty evaluation</td>
<td></td>
</tr>
</tbody>
</table>

Tab. 2 - Simplified list of the steps of the UMAE implementation, /1/.
<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>LOBI</th>
<th>SPES</th>
<th>BETHSY</th>
<th>LSTF</th>
</tr>
</thead>
<tbody>
<tr>
<td>PRZ Pressure</td>
<td>AA</td>
<td>0.03546</td>
<td>0.06898</td>
<td>0.09036</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.05094</td>
<td>0.06327</td>
<td>0.05229</td>
</tr>
<tr>
<td>SG pressure (secondary side)</td>
<td>AA</td>
<td>0.20812</td>
<td>0.12579</td>
<td>0.09656</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.04569</td>
<td>0.05468</td>
<td>0.05776</td>
</tr>
<tr>
<td>Heater rod temper. (high level)</td>
<td>AA</td>
<td>0.77903</td>
<td>0.58971</td>
<td>0.57610</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.05929</td>
<td>0.03693</td>
<td>0.05837</td>
</tr>
<tr>
<td>Primary side total mass</td>
<td>AA</td>
<td>0.34338</td>
<td>0.52439</td>
<td>0.14704</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.06833</td>
<td>0.06771</td>
<td>0.05060</td>
</tr>
<tr>
<td>DP inlet-outlet SG (IL)</td>
<td>AA</td>
<td>0.92310</td>
<td>1.15701</td>
<td>1.53348</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.05043</td>
<td>0.05075</td>
<td>0.05090</td>
</tr>
<tr>
<td>DP loop seal (descend. side BL)</td>
<td>AA</td>
<td>0.52696</td>
<td>0.59926</td>
<td>0.36891</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.07738</td>
<td>0.04273</td>
<td>0.08643</td>
</tr>
<tr>
<td>Overall accuracy (over 21 variables)</td>
<td>AA</td>
<td>0.333</td>
<td>0.320</td>
<td>0.278</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.055</td>
<td>0.055</td>
<td>0.056</td>
</tr>
</tbody>
</table>

Tab. 3 - Summary of results obtained for some relevant parameters by application of the FFT method to the selected parameters in the four counterpart experiments.

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>LOBI BL-34</th>
<th>SPES SP-SB-03</th>
<th>BETHSY 6.2 TC</th>
<th>LSTF scaled</th>
</tr>
</thead>
<tbody>
<tr>
<td>PRZ Pressure</td>
<td>AA</td>
<td>0.1002</td>
<td>0.108</td>
<td>0.0761</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.021</td>
<td>0.034</td>
<td>0.041</td>
</tr>
<tr>
<td>SG pressure (secondary side)</td>
<td>AA</td>
<td>0.1898</td>
<td>0.2640</td>
<td>0.0731</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.046</td>
<td>0.044</td>
<td>0.029</td>
</tr>
<tr>
<td>Heater rod temper. (high level)</td>
<td>AA</td>
<td>0.3705</td>
<td>0.2062</td>
<td>0.186</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.022</td>
<td>0.032</td>
<td>0.035</td>
</tr>
<tr>
<td>Primary side total mass</td>
<td>AA</td>
<td>0.2269</td>
<td>0.1659</td>
<td>0.177</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.061</td>
<td>0.065</td>
<td>0.075</td>
</tr>
<tr>
<td>DP loop seal (descend. side BL)</td>
<td>AA</td>
<td>0.4848</td>
<td>0.1859</td>
<td>0.4077</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.059</td>
<td>0.058</td>
<td>0.063</td>
</tr>
<tr>
<td>Overall accuracy (over 20 variables)</td>
<td>AA</td>
<td>0.3468</td>
<td>0.2975</td>
<td>0.2493</td>
</tr>
<tr>
<td></td>
<td>WF</td>
<td>0.04</td>
<td>0.048</td>
<td>0.048</td>
</tr>
</tbody>
</table>

Tab. 4 - Summary of results obtained for some relevant parameters by application of the FFT method to the selected parameters of the experimental counterpart data base and to the 'LSTF K_v scaled' calculation for the KRSKO plant.
<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>LOBI (ISP 18)</th>
<th>SPES (ISP 22)</th>
<th>BETHSY (ISP 27)</th>
<th>LSTF (ISP 26)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PRZ Pressure</td>
<td>AA 0.5166</td>
<td>0.2102</td>
<td>0.1068</td>
<td>0.0499</td>
</tr>
<tr>
<td></td>
<td>WF 0.107</td>
<td>0.03</td>
<td>0.033</td>
<td>0.027</td>
</tr>
<tr>
<td>Heater rod temperature</td>
<td>AA -</td>
<td>0.1199</td>
<td>0.4062</td>
<td>0.3346</td>
</tr>
<tr>
<td></td>
<td>WF -</td>
<td>0.024</td>
<td>0.03</td>
<td>0.055</td>
</tr>
<tr>
<td>Primary side</td>
<td>AA 0.4182</td>
<td>-</td>
<td>-</td>
<td>0.1174</td>
</tr>
<tr>
<td>total mass</td>
<td>WF 0.059</td>
<td>-</td>
<td>-</td>
<td>0.071</td>
</tr>
<tr>
<td>AA_{tot} (tot. 4 variables)</td>
<td>0.401</td>
<td>0.372</td>
<td>0.389</td>
<td>0.156</td>
</tr>
<tr>
<td>WF_{tot}</td>
<td>0.099</td>
<td>0.058</td>
<td>0.078</td>
<td>0.047</td>
</tr>
</tbody>
</table>

(*) : In some case, data were missing.

**Tab. 5** - Summary of results obtained for some relevant parameters by application of the FFT method in order to verify the qualification of the nodalizations set up for LOBI/MOD2, SPES, BETHSY, LSTF facilities.

**Fig. 1** - Summary of the sample rod cladding temperature trends ("experimental" and calculated) considered for the evaluation of the correlation between AA and |PCT_{cal}-PCT_{exp}|.
Fig. 2 - Summary of results obtained for the parametric analysis of the AA dependence from the $|PCT_{cal}-PCT_{exp}|$, performed for a typical Intermediate Break LOCA cladding temperature trend.
OVERVIEW OF UMAE-SETF
(Extension at the Separate Effects Test Facilities of the Uncertainty Methodology based on Accuracy Extrapolation)
AND QUALIFICATION PROCESS

(""")AKSAN S.N
(""")D'AURIA F.
(""")FALUOMI V

(*Department of Nuclear and Mechanical Constructions
Pisa University
Italy

(**)Department of Thermohydraulic
Paul Scherrer Institute
Switzerland

CSNI UNCERTAINTY WORKSHOP, LONDON,
MARCH 1-4, 1994

[A] Objective of UMAE-SETF and Application

[B] General Description of UMAE-SETF
- Flow sheet of UMAE with SETF extensions
- Steps of the UMAE-SETF
- Particularities of methodology
- Results of extrapolation of accuracy

[C] Qualification Process and Results
- Main characteristics of LSTF facility
- Experimental test considered and calculation results
- Phenomena plane and phenomenological areas
- Results of qualification process

[D] Conclusions
[B] GENERAL DESCRIPTION OF UMAE-SETF

B.1) Flow sheet of UMAE with SETF extensions
B.2) Steps of the UMAE-SETF

1. Identification of thermal hydraulic phenomenon to analyse.

2. Definition of an experimental data base through the identification of suitable facilities and experimental tests.

3. Similitude analysis of the experimental data and "grouping" of homogeneous points.

4. Use of a qualified code to get the calculated data base related to the considered tests.

5. Qualitative accuracy of calculated data. Quantitative accuracy is possible, but not performed so far.

6. Extrapolation of accuracy using the statistical methodology (already used in the UMAE) and calculation of uncertainty for the single phenomenon.
B.3) Particularities of methodology

Identification of phenomenon:

The phenomenon was identified using a set of characteristics events, called relevant events.

Each event is identified by a couple of values, i.e. cladding temperature and time of occurrence.

It was defined also three different slopes of cladding temperature curve, used for detecting the different kinds of heat exchange regimes during the transient.

The relevant events detected for a generic dry-out condition are shown in the following figure:

![Diagram showing temperature-time relationship with identified events]

In the next table are reported the main characteristics of design of above cores:

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>UNIT</th>
<th>NEPTUN</th>
<th>THTF</th>
<th>LOHI</th>
<th>REFERENCE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Volume of fluid in the core</td>
<td>m³</td>
<td>0.017</td>
<td>0.038</td>
<td>0.062</td>
<td>90.544</td>
</tr>
<tr>
<td>Pressure</td>
<td>MPa</td>
<td>0.30</td>
<td>14.200</td>
<td>15.7</td>
<td>15.6</td>
</tr>
<tr>
<td>Power</td>
<td>MW</td>
<td>0.076</td>
<td>15.00</td>
<td>15.0</td>
<td>15.0</td>
</tr>
<tr>
<td>Number of rods</td>
<td></td>
<td>25</td>
<td>60</td>
<td>64</td>
<td>28453</td>
</tr>
<tr>
<td>Lattice</td>
<td></td>
<td>square</td>
<td>square</td>
<td>square</td>
<td>square</td>
</tr>
<tr>
<td>Pitch</td>
<td>m</td>
<td>0.014</td>
<td>0.014</td>
<td>0.014</td>
<td>0.013</td>
</tr>
<tr>
<td>D₄ core</td>
<td>m</td>
<td>0.10</td>
<td>0.011</td>
<td>0.012</td>
<td>0.011</td>
</tr>
<tr>
<td>D₄ central</td>
<td>m</td>
<td>0.011</td>
<td>0.014</td>
<td>0.015</td>
<td>0.010</td>
</tr>
<tr>
<td>D₄ outer</td>
<td>m</td>
<td>0.100</td>
<td>0.147</td>
<td>0.122</td>
<td>0.170</td>
</tr>
<tr>
<td>Heated length</td>
<td>m</td>
<td>1.990</td>
<td>3.880</td>
<td>2.6</td>
<td>3.880</td>
</tr>
<tr>
<td>Flow area core</td>
<td>m²</td>
<td>0.004</td>
<td>0.004</td>
<td>0.008</td>
<td>0.014</td>
</tr>
<tr>
<td>Active heat exchange area</td>
<td>m²</td>
<td>8.66</td>
<td>2.552</td>
<td>2.275</td>
<td>3.216</td>
</tr>
<tr>
<td>Elevation of inlet water</td>
<td>m</td>
<td>0.447</td>
<td>0.06</td>
<td>1.83</td>
<td></td>
</tr>
<tr>
<td>Elevation of outlet water</td>
<td>m</td>
<td>1.530</td>
<td>1.000</td>
<td>1.770</td>
<td></td>
</tr>
<tr>
<td>Reference elevation of TC</td>
<td>m</td>
<td>0.947</td>
<td>3.25</td>
<td>2.13</td>
<td></td>
</tr>
</tbody>
</table>

* The THTF power shape is flat.
For analyzing the similitude among the data obtained from the different tests, the following steps are used:

- An X-Y plane was chosen for collapsing the information about the dry-out phenomenon (phenomena plane). The X and Y variables were chosen to be function of:
  - the boundary conditions of phenomenon (pressure, mass flow, etc.);
  - the design of the facilities considered;
  - the initial conditions of the tests considered.

- For each point in the X-Y plane "grouping bands" including the "definition error" in the position of points are defined.

- The intersection of the "grouping bands" brings to the "phenomenological areas"; homogeneous phenomena are suppose to be found inside each "phenomenological area".

\[
Y(X) = \left( \frac{4 \cdot L_h \cdot A_{FAC} \cdot q_L (X) - 1}{d_h \cdot A_{AME} \cdot q_a} \right)
\]

and X is the ratio between the phase change number and subcooling number:

\[
X = \frac{\dot{Q}}{G \cdot c_p \cdot \Delta T_{SUB}}
\]

- The "definition errors" in the position of each point in the plane was considered as the differences from the predicted value for the reference plant of the value obtained in the facility (for both X and Y quantities):

\[
Y_{REF} = Y_{FAC} \pm \Delta Y
\]

\[
\Delta Y = f(\text{design parameters, initial and boundary conditions})
\]

where K and W are scaling factors connecting the parameter Y measured in the facility and predicted in the plant.
General description of UMAE-SETF

Experimental data base definition: experimental tests

Trend of cladding temperature for the NEPTUN boil-off tests

Trend of cladding temperature for the THTF film boiling tests

Trend of cladding temperature for the LOBI SB-LOCA test

7/20
Example of phenomena plane for the PCT event

General description of UMAE-SETF

The phenomenological area contains the points 1, 2, 3

phenomenological area
General description of UMAE-SETF

**Calculated data base:** comparison between cladding temperatures

[Graph showing temperature over time for NEPTUN 5011 test]

Calculated and measured cladding temperature for the NEPTUN 5011 test

[Graph showing temperature over time for THTF 3.03.6AR test]

Calculated and measured cladding temperature for the THTF 3.03.6AR test

[Graph showing temperature over time for LOBI BL-34 test]

Calculated and measured cladding temperature for the LOBI BL-34 test

11/20
Accuracy extrapolation: basic methodology

- the average accuracy is defined as the average of distance from the unit of the ratio between experimental and calculated data:

\[
\bar{A} = \frac{\sum_{i=1}^{N_{PA}} \left( \frac{Y_e}{Y_C} \right)_i - 1}{N_{PA}}
\]

- 95% accuracy:

\[(A)_{95} = \bar{A} + 2 \sigma\]

- value for the reference plant:

\[Y_R = \frac{\sum_{i=1}^{N_{PA}} Y_{e_i}}{N_{PA}}\]

- uncertainty:

\[\bar{U} = \bar{A} \cdot Y_R\]

- 95% uncertainty:

\[(U)_{95} = (A)_{95} \cdot Y_R\]

---

General description of UMAE-SETF

Accuracy extrapolation: refined methodology

"Weights" for the experimental data are introduced considering that larger facilities represent better the plant behaviour.

The weights are defined analysing the parameters that give highest distortions in the representation of plant behaviour in the core of facilities:

1. active heat exchange area of the channel;
2. flow area of the channel;
3. fluid energy of the channel.

The final weight, called \(P_i\), it is the average of the previous three ones:

\[
\bar{A} = \frac{\sum_{i=1}^{N_{PA}} \left( \frac{Y_e}{Y_C} \right)_i - 1}{N_{PA}}
\]

A global error related to the dimensions of the facilities is also introduced:

\[E = \left| \frac{0.3}{\sqrt{K_V}} \right|\]

The aforementioned error it is used in the uncertainty calculation:

\[\bar{U} = \bar{A} \cdot Y_R \cdot (1 + E)\]

\[(U)_{95} = [\bar{A} \cdot Y_R + 2 \cdot \sigma_{Y_R} + 2 \cdot \sigma_{\bar{A}} \cdot \bar{A}] \cdot (1 + E)\]
### General description of UMAE-SETF

#### B.4) Results of accuracy extrapolation.

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>BASIC</th>
<th>METHOD</th>
<th>Refined</th>
<th>METHOD</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>A</td>
<td>U</td>
<td>A</td>
<td>U</td>
</tr>
<tr>
<td>Time of CHF (s)</td>
<td>41%</td>
<td>16%</td>
<td>41%</td>
<td>12%</td>
</tr>
<tr>
<td>Time of HU (s)</td>
<td>41%</td>
<td>18%</td>
<td>12%</td>
<td>10%</td>
</tr>
<tr>
<td>Time of GC (s)</td>
<td>23%</td>
<td>12%</td>
<td>23%</td>
<td>12%</td>
</tr>
<tr>
<td>Time of PCT (s)</td>
<td>7%</td>
<td>3%</td>
<td>7%</td>
<td>3%</td>
</tr>
</tbody>
</table>

#### Whole database

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>BASIC</th>
<th>METHOD</th>
<th>Refined</th>
<th>METHOD</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>A</td>
<td>U</td>
<td>A</td>
<td>U</td>
</tr>
<tr>
<td>Steps of HU (s)</td>
<td>39%</td>
<td>11%</td>
<td>39%</td>
<td>11%</td>
</tr>
<tr>
<td>Steps of PCT (s)</td>
<td>43%</td>
<td>19%</td>
<td>43%</td>
<td>19%</td>
</tr>
</tbody>
</table>

#### Phenomenological area E

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>BASIC</th>
<th>METHOD</th>
<th>Refined</th>
<th>METHOD</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>A</td>
<td>U</td>
<td>A</td>
<td>U</td>
</tr>
<tr>
<td>Steps of HU (s)</td>
<td>27%</td>
<td>13%</td>
<td>27%</td>
<td>13%</td>
</tr>
<tr>
<td>Steps of PCT (s)</td>
<td>27%</td>
<td>13%</td>
<td>27%</td>
<td>13%</td>
</tr>
</tbody>
</table>

---

### Qualification Process and Results

**[C] QUALIFICATION PROCESS AND RESULTS**

The qualification process is needed for having an indication about the capabilities of a methodology developed: it consists of the following steps:

a) choice of a larger facility than those considered in the experimental data base, suitable to be considered as a "reference plant".

b) choice of an experiment in which the phenomenon considered for the accuracy extrapolation occurs;

c) calculation of the scenario considered with the same code used for getting the data base of the methodology;

d) extrapolation of accuracy calculation; the results are related to the facility taken as reference plant;

e) calculation of the position, in the phenomena plane, of the phenomenon detected in the experimental scenario;

f) comparison between the uncertainty calculated for the relevant phenomenological areas and the error given by the code, obtained comparing calculated and experimental data.

---

This is the most representative value obtained from the UMAE-SETF.
Qualification Process and Results

C.1) Main characteristics of LSTF facility

The LSTF facility is a scale model of a Westinghouse type PWR (3423 MWT, 4 loops) with a volume scaling ratio equal to 1/48.

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>UNIT</th>
<th>LSTF</th>
</tr>
</thead>
<tbody>
<tr>
<td>VOLUME CORE</td>
<td>m³</td>
<td>0.407</td>
</tr>
<tr>
<td>Dh DOWNCOMER</td>
<td>m</td>
<td>0.100</td>
</tr>
<tr>
<td>HEIGHT UPPER PLENUM</td>
<td>m</td>
<td>4.940</td>
</tr>
<tr>
<td>HEIGHT LOWER PLENUM</td>
<td>m</td>
<td>2.260</td>
</tr>
<tr>
<td>PRESSURE</td>
<td>MPa</td>
<td>15.4</td>
</tr>
<tr>
<td>POWER</td>
<td>MWth</td>
<td>10</td>
</tr>
<tr>
<td>N.RODS</td>
<td></td>
<td>1008</td>
</tr>
<tr>
<td>Dc CORE</td>
<td>m</td>
<td>0.012</td>
</tr>
<tr>
<td>HEATED LENGTH</td>
<td>m</td>
<td>3.600</td>
</tr>
<tr>
<td>FLOW AREA CORE</td>
<td>m²</td>
<td>0.113</td>
</tr>
<tr>
<td>AREA EXCHANGE CORE</td>
<td>m²</td>
<td>110.107</td>
</tr>
<tr>
<td>PEAK FACTOR</td>
<td></td>
<td>1.550</td>
</tr>
</tbody>
</table>

Main characteristics of LSTF

---

C.2) Experimental test considered and calculation results

The SB-CL-18 test is essentially a small break LOCA originated by a rupture in the cold leg between the pump and the vessel. The sequence of interventions of the various systems is typical of the kind of transient in a plant: after the break, scram and pumps trip are provided together with a signal for isolating steam generators (feedwater and steam line closure).

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>UNIT</th>
<th>LSTF</th>
<th>REFERENCE</th>
</tr>
</thead>
<tbody>
<tr>
<td>PRESSURE</td>
<td>MPa</td>
<td>1.57E+01</td>
<td>1.57E+01</td>
</tr>
<tr>
<td>CORE MASS FLOW OUTLET</td>
<td>Kgs</td>
<td>1.89E+02</td>
<td>2.16E+03</td>
</tr>
<tr>
<td>SUBCOOLING FLUID CORE INLET</td>
<td>K</td>
<td>6.80E+01</td>
<td>7.10E+01</td>
</tr>
<tr>
<td>FLUID TEMPERATURE AT THE CORE OUTLET</td>
<td>K</td>
<td>5.67E+02</td>
<td>6.81E+02</td>
</tr>
<tr>
<td>POWER</td>
<td>KW</td>
<td>1.00E+04</td>
<td>1.82E+06</td>
</tr>
<tr>
<td>Number of Heaters</td>
<td></td>
<td>4.58E-01</td>
<td>4.93E-01</td>
</tr>
<tr>
<td>nhc</td>
<td></td>
<td>6.23E-01</td>
<td>8.34E-01</td>
</tr>
</tbody>
</table>

Initial conditions for the SB-CL-18 test and PWR operational conditions

LSTF SB-CL-18 test: experimental trends of main parameters

LSTF SB-CL-18 test: calculated and experimental trends of rod temperature
C.3) Phenomena plane and phenomenological areas

Final definition of phenomenological areas using the LSTF as reference plant

Position of the LSTF events in the phenomena plane

C.4) Results of qualification process

Accuracy extrapolation results with the LSTF as reference plant

Comparison between experimental data and predicted data with UMAE-SETF
It was investigated the possibility to analyse a single phenomenon using a set of experimental data from SETF and ITF, and it was shown that is reliable only using a method for grouping the data to get a homogeneous groups with which extrapolating the accuracy of code calculation.

The UMAE-SETF largely improves the range of applicability of UMAE: in fact, relevant data from SETF can be exploited.

The considered application lead to an error of ±269 K in the prediction of the PCT in the plant when the phenomenon falls inside the specified phenomenological area.

The same error holds ±216 K for predicting of PCT in LSTF. The range resulting from UMAE-SETF for the PCT is 712 ±216.

The PCT in the experiment performed in the LSTF is 608 K, falling inside the uncertainty bands.
AEA RS 5522

A REVIEW OF
LOCA UNCERTAINTY STUDIES IN AEA

A. J. Wickett
June 1993
Reactor Safety Studies Department
Safety and Performance Division
AEA Reactor Services
Winfirth Technology Centre

Presented at
Joint International Conference on
Mathematical Models and Supercomputing in Nuclear Applications
Karlsruhe, Germany, 19-23 April 1993

HSE SPONSORED NUCLEAR SAFETY RESEARCH PROGRAMME
PWR Sub-Programme
AEA Project No: LHT 1.4
HSE NSRMU Project No: HTH 004

AEA Reactor Services,
Winfirth Technology Centre,
Dorchester,
Dorset
DT2 8DH
UK
RECORD OF INTERNAL REVIEW FOR AEA RS 5522

<table>
<thead>
<tr>
<th>Reviewed and Approved by</th>
<th>Name</th>
<th>Signature</th>
<th>Position/Organisation</th>
<th>Date</th>
</tr>
</thead>
<tbody>
<tr>
<td>Author</td>
<td>Dr A. J. Wickett</td>
<td><img src="image" alt="Signature" /></td>
<td>RS/SAPD/RSSD</td>
<td>23/6/93</td>
</tr>
<tr>
<td>Peer Reviewer</td>
<td>Dr J. N. Lillington</td>
<td><img src="image" alt="Signature" /></td>
<td>Project Manager, RS/SAPD/RSSD</td>
<td>23/6/93</td>
</tr>
<tr>
<td>Approver</td>
<td>Dr S. R. Kinnersly</td>
<td><img src="image" alt="Signature" /></td>
<td>Department Manager, RS/SAPD/RSSD</td>
<td>24/6/93</td>
</tr>
</tbody>
</table>
A REVIEW OF LOCA UNCERTAINTY STUDIES IN AEA

A. J. Wickett

AEA Reactor Services, Winfrith Technology Centre

SUMMARY

The purpose of this report is to review and summarise the AEA's achievements in the area of LOCA uncertainty analysis since 1986 so as to make recommendations for the next steps in this work.

This report therefore describes the AEA methodology for evaluating the uncertainty in calculations of PWR LOCAs and two studies to demonstrate it. Because of various difficulties in modelling LOCA the uncertainty analysis is also difficult and the methodology had to be justified appropriately from first principles. The starting point was that reliable advice is needed by people who have to take decisions about the design, siting, licencing and safe operation of plant.

The "AEA method" for uncertainty analysis consists of three elements:
- definition of a "reasonable uncertainty range"
- method of combining uncertainties and
- a flow chart for organising full scale plant analysis.

Parts of the method have been demonstrated in two sets of calculations, for the LOBI test BL-02 and for a PWR large LOCA. For these calculations both TRAC-PFI/MOD1 (for the LOBI calculation) and TRAC-PF1/MOD2 (for the PWR) were modified to allow modelling parameters to be introduced to physical subroutines from input.

For the LOBI test 20 calculations with TRAC-PFI/MOD1 were successfully run and reported. The predicted uncertainty ranges mostly bounded the data. The large LOCA calculations used TRAC-PF1/MOD2 and had to be curtailed due to poor mass conservation. However 26 calculations of the first 50 s of the transient were accomplished.

The two studies reported have revealed no difficulties in operating the AEA method that cannot be overcome in practice and no problems that invalidate the method. The LOBI calculations show that uncertainty ranges for calculated quantities can be successfully calculated. The AEA method is therefore recommended for further use.

The appropriate next steps are identified.
<table>
<thead>
<tr>
<th>CONTENTS LIST</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Title Page</strong> ..........................................................</td>
</tr>
<tr>
<td><strong>Record of Review</strong> ..................................................</td>
</tr>
<tr>
<td><strong>Summary</strong> ..................................................................</td>
</tr>
<tr>
<td><strong>Contents</strong> ...............................................................</td>
</tr>
<tr>
<td><strong>List of Figures</strong> .....................................................</td>
</tr>
<tr>
<td><strong>1. Introduction</strong> ......................................................</td>
</tr>
<tr>
<td><strong>2. The AEA Method</strong> ..................................................</td>
</tr>
<tr>
<td>2.1. Definition of Uncertainty Range ..................................</td>
</tr>
<tr>
<td>2.2. Use of Subjective Probabilities ..................................</td>
</tr>
<tr>
<td>2.3. Characterisation of the LOCA Problem ..........................</td>
</tr>
<tr>
<td>2.4. Combination of Uncertainties ....................................</td>
</tr>
<tr>
<td>2.5. Flow Chart for Plant Calculations ...............................</td>
</tr>
<tr>
<td><strong>3. LOBI BL-02 Pilot Study</strong> .........................................</td>
</tr>
<tr>
<td><strong>4. Large LOCA Study</strong> ..................................................</td>
</tr>
<tr>
<td><strong>5. Other Uncertainty Analysis Methods</strong> ..........................</td>
</tr>
<tr>
<td>5.1. CSAU ..................................................................</td>
</tr>
<tr>
<td>5.2. GRS Method ...........................................................</td>
</tr>
<tr>
<td><strong>6. Conclusions and Recommendations</strong> ...............................</td>
</tr>
<tr>
<td>6.1. Conclusions ...........................................................</td>
</tr>
<tr>
<td>6.2. Recommendations ........................................................</td>
</tr>
<tr>
<td><strong>7. References</strong> ...........................................................</td>
</tr>
<tr>
<td><strong>Table 1: Blowdown and Post-Blowdown PCTs for Single Variations</strong></td>
</tr>
<tr>
<td><strong>Table 2: Blowdown and Post-Blowdown PCTs for Multiple Variations</strong></td>
</tr>
</tbody>
</table>

**List of Figures**

| Figure 1 | Flow Chart for Uncertainty Prediction for Reactor. From Reference 1. | 21 |
| Figure 2 | LOBI BL-02 Experiment. Rig Data and Calculated Range, Total Primary System Mass. From Reference 1. | 22 |
| Figure 3 | 4 Loop Westinghouse PWR. Max Temp of Hot Rod 2. Base Case (BC2) and Upper and Lower PCTs (1a, 2b, 2c). From Reference 6. | 23 |

**AEA RS 5522**
1. Introduction

Theoretical and experimental studies of loss of coolant accidents (LOCAs) of pressurised water reactors (PWRs) have been carried out by AEA since the early 1980s. Limitations of experimental data and uncertainty in modelling make the prediction of such accidents uncertain. Projects to investigate the uncertainty in such predictions have been pursued by AEA since 1986. These projects were funded by the then Central Electricity Generating Board and the Department of Energy and currently by the Health and Safety Executive.

The purpose of this report is to review and summarise the AEA’s achievements in this area so as to make recommendations for the next steps.

The estimation of uncertainty in LOCA calculations is difficult for a variety of reasons which will be elaborated in Section 2.3. It therefore is a rigorous test of uncertainty estimation methodologies. For this reason the first step we took was to choose an uncertainty analysis methodology. This choice was based upon the requirement to provide useful information to decision makers. The "AEA method" is described and justified in Section 2 of this report. It comprises three main elements:

- definition of a "reasonable uncertainty range"
- method of combining uncertainties and
- a flow chart for organising full scale plant analysis.

It was decided at an early stage to demonstrate the method in the analysis of an integral experiment. The small break LOCA test LOBI BL-02 was chosen. This analysis is described in Section 3 of this report. Following the successful LOBI calculations a study of the uncertainty in full-scale plant predictions of a large LOCA is in progress. This is described in Section 4.

The importance of uncertainty analysis for LOCA calculations has been generally recognised in countries operating PWRs. Other analysis methods have been devised in the United States and are under development in Germany and France. This report is not intended to be a comparative review. However two of these other methods are briefly mentioned in Section 5. The international interest in this subject means that it is currently under discussion by the OECD NEA CSNI. It also led to an invitation to present this paper at the Joint International Conference on Mathematical Models and Supercomputing in Nuclear Applications in Karlsruhe, Germany in April 1993.

Conclusions and recommendations are given in Section 6 of this report.
2. The AEA Method

2.1. Definition of Uncertainty Range

The starting point in choosing how to characterise uncertainty in the result of a predictive reactor safety analysis, such as a LOCA study, is that the results should be expressed so as to be as useful as possible to anyone who is to base decisions about actions on them. Such decisions might relate to the design, siting, licencing and safe operation of the plant. This means stating which values of the quantity of interest are consistent with the evidence. Any of these values might be the (unknown) true value. A wider statement of uncertainty would not use all the evidence and would include some possibilities that should be excluded from consideration by the decision-maker. A narrower statement of the uncertainty would omit some possibilities that were consistent with evidence. This would have the undesirable consequence that the decision-maker would not consider values that were consistent with evidence. This could mean that in the event of a LOCA in a real plant the progress of the accident would be outside the range of descriptions that were considered when deciding on the design of the plant. Thus public health and safety would not be ensured.

If these considerations are applied to the uncertainty in continuous variables such as temperature, they lead to the definition of a reasonable uncertainty range 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.

Since mathematical proof of the inconsistency is not possible or required the standard of "reasonable certainty" is adopted. This means that reasoning, sufficient to satisfy peer reviewers knowledgeable in the area, must be given to justify the inconsistency. There are two cases:

The first we call the "data based" case. If a single constant quantity is measured repeatedly then a best estimate and a classical statistical 95% confidence interval can be derived. Which interval is chosen, 95% for example, gives the degree to which values outside the interval are inconsistent with the evidence of the measurements.

However, many of the quantities in LOCA analysis have not been measured and these we call "data free". In such cases other physical arguments have to be put forward to convey reasonable certainty to the peer reviewer that the values outside the range are inconsistent with the available evidence.

These arguments are set out in some more detail in Section 2 of Reference 1.

2.2. Use of Subjective Probabilities

Some analysts use their personal subjective probability distributions of uncertain quantities to describe uncertainty ranges, even when the quantities are not random quantities. Such distributions were, for example, used in the USNRC's Code Scaling, Applicability and Uncertainty
methodology, which will be discussed further in Section 5. Such distributions would almost
certainly be changed by subsequent evidence. The analyst's subjective probability density
would increase in part of the range. It is therefore possible that the subjective probability of
part of the range would increase so as to bring into consideration by the decision maker
values that had previously been ignored because of low subjective probability.

Such characterisation of uncertainty ranges does not, therefore, meet the requirement that they
are useful to decision makers.

2.3. Characterisation of the LOCA Problem

The modelling and data used to describe a LOCA have several characteristics which deter-
mine how the uncertainty analysis must be performed. First we consider the modelling:
- A substantial number of processes in heat transfer, two-phase flow and other areas are
  modelled.
- Some of the models are highly non-linear and some have discontinuities and cliff-edges.
- Some of the models are complex and some interact with one another in complex ways.
- The models are embodied in a large computer code (such as TRAC) which is expensive
to run (~ 5 hours on the CRAY-2 using TRAC-PF1/MOD2).

Identification of uncertainties reveals that:
- Uncertainties exist in models, data and initial and boundary conditions.
- The data sources are diverse but incomplete, so some extrapolation is needed.
- Uncertainties are mostly of the data free kind (defined in Section 2.1).
- Many (~100) uncertainties can be identified. When these are narrowed down to the im-
  portant ones, several (~10) still remain.

The implications of these characteristics of the problem for uncertainty analysis are discussed
in Sections 2.4 and 2.5.

2.4. Combination of Uncertainties

We use the term "uncertainty space" to mean the space of values defined by reasonable uncer-
tainty ranges for all the uncertain quantities - input, data, initial and boundary conditions and
model parameters - in the problem.

In Section 2.3 we mentioned that a large number of uncertainties have to be narrowed down
to a smaller number of important uncertainties for inclusion in the calculations. The analyst
must exercise judgement informed by all available information for this. Preliminary calcula-
tions, including hand calculations, may be used. It may be valuable to review the uncertain-
ties omitted after the "important" ones have been combined to see whether new insights into
the problem suggest that another uncertainty might be important.
Because most uncertainties are data-free, it follows from the definition of "reasonable uncertainty range" that the reasonable uncertainty range for the quantity of interest being calculated - usually peak clad temperature (PCT) for a large LOCA - is obtained by maximising and minimising the quantity of interest over the uncertainty space.

The uncertainty space will have a number of dimensions (say, 10). Because calculations are expensive it has to be sampled sparsely. Systematic sampling (for example latin hypercube sampling) is therefore not possible.

The PCT will vary in a non-monotonic, non-linear and discontinuous way in various parts of the uncertainty space. It cannot, therefore, be approximated by a polynomial response surface over the whole space.

The uncertain quantities are not (mostly) random variables. Therefore the arguments set out in Section 2.2 mean that a probability density over the uncertainty space will not be available.

It follows that the most cost-effective way of sampling in the uncertainty space is to obtain the best possible understanding of the problem and the interactions of the various processes and to use scientific and mathematical judgement iteratively to seek the overall maximum and minimum.

2.5. Flow Chart for Plant Calculations

Because the modelling and data used in the TRAC code comes from a wide diversity of sources and extrapolation from these sources to the plant case is needed; and because of the difficulty, outlined in Section 2.4, of the overall maximisation and minimisation process, the flow chart set out in Figure 1 is proposed. This sets out the organisation of the flow of information from separate effect and integral experiments to the plant predictions.

The important feature of this chart is that some integral data are kept separate from the data base used to develop, refine and test the code. 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 two or more compensating errors. Since a large LOCA is outside the development data base for TRAC, it is the ability of the code to predict uncertainty ranges outside the development data base that is of interest. After the development loop is completed - in which the code and the techniques for maximising and minimising over the uncertainty space are refined so that uncertainty ranges for development data are narrowed - the code is used to predict uncertainty ranges for data from the independent data base. If these data are encompassed by the predicted uncertainty ranges the method is validated. However if the data fall outside the ranges then further work is needed followed by a new comparison of predicted uncertainty ranges with new independent data.

The flow chart is further explained and elaborated in Section 2.4 of Reference 1.
3. LOBI BL-02 Pilot Study

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 small break LOCA experiment BL-02, was made. The uncertainty study was conducted "blind", that is the results of the experiment were not examined until all the calculations had been made and the uncertainty ranges predicted. The code used was TRAC-PF1/MOD1 version 13.0 plus generalised heat structures modified to allow models to be varied by means of parameter values obtained from input.

This pilot study is described fully in Reference 1.

Modelling and input uncertainties that were identified as important and were implemented in the pilot study were:
- interphase friction
- interfacial heat transfer
- choked flow and
- bypass flow area.

Other uncertainties, judged to be important but not implemented in the pilot study, were counter-current flow limitation and offtake modelling.

In this study, not one but four calculated quantities were identified to have their uncertainty estimated. They were:
- primary system coolant mass (as a function of time)
- coolant mass in the core (as a function of time)
- minimum coolant mass in the core
- the time of that minimum.

The first two of these are functions of time. To facilitate bounding calculations they were replaced by discrete quantities. The study was limited to the period up to accumulator discharge (≈ 650 s).

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. Then a further six calculations in which the uncertainties were varied together were run. When the resulting calculated uncertainty ranges were compared with the data from the rig, the two functions of time were mostly bounded and the two discrete quantities were bounded by the calculated ranges. Figure 2 compares the calculated uncertainty range for total primary system mass with the measured values. The failures to bound data were attributed to an uncertainty that had been shortlisted as important but not included in the study for reasons of time (offtake modelling) and an uncertainty that was possibly understated (choked flow modelling).
It was concluded that the method meets the requirements for an uncertainty study.
4. Large LOCA Study

This study investigates the uncertainty in peak clad temperature (PCT) in a large LOCA calculation for a 4 loop Westinghouse PWR. The calculations used TRAC-PF1/MOD2 version 5.3. This code version was chosen for speed, so that the number of calculations needed could be carried out.

Initially we identified and ranged uncertainties relevant to modelling the large LOCA. These uncertainties were quantified by reference to TRAC-PF1/MOD1 modelling (where necessary) because documentation for the faster, MOD2, code version was not available at the time. These results were reported in Reference 2.

When TRAC-PF1/MOD2 and its documentation became available, work to modify MOD2 so that uncertain parameters could be varied through input was carried out. This is reported in reference 3. Up to now, most, but not all, of the important uncertainties have been implemented in MOD2. Reference 3 also describes choices made for the base case in the present study.

Out of the important uncertainties identified in reference 2 the following were implemented for this part of the study [3]:

Selection of scenario (in part - 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
Effect of dissolved nitrogen in ECCS.

Not yet implemented are:

Interphase heat transfer
Wall heat transfer - MOD2 reflood model
Some aspects of selection of scenario - primary pressure peaks, SGTR
Moderator feedback.

Initial base case calculations with TRAC-PF1/MOD2 showed significant differences from typical calculations with TRAC-PF1/MOD1 (for example reference 4). These differences are described in Reference 5. An important difference was that the version of TRAC-PF1/MOD2 used (version 5.3 modified to allow uncertain model parameters to be varied through input) had poor mass conservation. For this reason only the first 50 s of the transient was studied.

During the first 50 s the behaviour of the maximum clad temperature (ie the highest calculated clad temperature, maximised over the core, as a function of time) can be described in three phases. The first is the blowdown phase during which the blowdown peak temperature typically occurs at \( \approx 5 \) s. The core is then cooled by water entering from top and bottom. The lower plenum fills from \( \approx 20 \) s to \( \approx 40 \) s. This is the refill phase. The core then fills during the reflood phase. For the purposes of this study we refer to the refill phase and the reflood
phase up to 50 s as the "post-blowdown" period. During this time there is typically another temperature peak. We now define more precisely the objectives of this part of the study as to maximise and minimise the blowdown and the post-blowdown peak clad temperatures (PCTs).

A base case and 19 calculations in which one uncertain quantity was varied at a time were run. The results are set out in Table 1. This is extracted from Reference 6. PCTs are calculated for two "hot rods", HR1 and HR2. HR1 is intended to provide the reasonable upper limit for the total peaking factor justified in References 2 and 3. This total peaking factor is 1.7 near the beginning of the reactor life, but varies with time in life. The other hot rod, HR2, gives a total peaking factor of 2.32. This corresponds to the design limit for this quantity. Although this value is outside the reasonable range so far derived for this parameter, it was included at the request of HM Nuclear Installations Inspectorate and is widely used in limiting design basis large LOCA calculations.

The effects of the single variations were summarised as follows:

- Large impacts:
  1. Time in reactor life/fuel state
  2. Post-CHF Heat transfer (post-blowdown only)

- Medium impacts:
  1. Break flow (most significantly post-blowdown)
  2. Bubbly-slug interphase drag (blowdown only)
  3. Forced convection heat transfer
  4. Minimum film boiling temperature

- Small impacts:
  1. Cladding conductivity
  2. Entrained droplet size
  3. Entrainment fraction
  4. Stratified interphase drag

Based on the results of the single variations, six calculations were made in which all the uncertainties were varied to maximise and minimise PCT. The results are set out in Table 2. Cases 1a and 2a are based on a straightforward examination of Table 1 (set out in detail in Table 2 of Reference 6). Two of the uncertainties were chosen for further examination because of possible non-monotonic behaviour and/or synergisms. The two single variations about case 1a, which aims to maximise PCT use low break flow rather than high break flow (case 1b), and minimum $T_{\text{min}}$ instead of maximum $T_{\text{min}}$ (case 1c). The two single variations about case 2a, which aims to minimise PCT use low break flow rather than standard break flow (case 2b), and maximum $T_{\text{min}}$ instead of minimum $T_{\text{min}}$ (case 2c).
It was necessary to complete this phase of the calculations within a limited time. Otherwise, more calculations would have been performed. Figure 3 shows a plot of the HR2 maximum temperature against time for the base case and cases 1a (maximum) and 2b and 2c (minimum).

The overall ranges for PCT were: for the blowdown peak clad temperature, a range of 737 – 1042 K for the hot rod with a reasonable upper limit peaking factor, and a range of 909 – 1221 K for the second hot rod (total peaking factor = 2.32). The post-blowdown peak clad temperature ranges were 548 – 1016 K and 601 – 1232 K for the two hot rods respectively. These uncertainty ranges should be compared with the USNRC "Appendix K" limit of 1477 K and the temperatures required for extensive clad ballooning, in the region of 1200 K.

The poor mass conservation with the code version used means that these calculations should be revisited and run to cover the whole transient when the problem has been cured.
5. Other Uncertainty Analysis Methods

It is not the purpose of this report to give an exhaustive and comparative review of the other methods proposed or under development for estimating the uncertainty in LOCA calculations. One review has been published and another is in preparation [7, 8]. Our abbreviated summaries of two of the other methods follow.

5.1. CSAU

The USNRC’s Code Scaling, Applicability and Uncertainty methodology systematically identifies and ranges uncertainties. TRAC calculations rely on "supplemental rods" to enhance the effective number of calculations. The variation of PCT over the uncertainty space is then fitted with a polynomial response surface. Subjective probability distributions are assigned to input uncertainties. These imply a subjective probability density over the uncertainty space. The response surface is used instead of TRAC calculations in a Monte-Carlo study in which the uncertainty space is sampled at random according to the subjective probability density. The 5th and 95th percentiles of the resulting subjective probability distribution of PCT are noted. These are combined with other uncertainties, not included in the main study, by additive "biases". There is no test against independent data [9, 10].

AEA editors prepared the UK’s review of the CSAU study [11].

5.2. GRS Method

A subjective probability density distribution is set up over the uncertainty space and sampled by a fixed number of calculations. The number of calculations is based on a statistical analysis and depends on the subjective confidence level required. Thus for a 95% probability and 95% subjective confidence level, 59 calculations are said to be needed to bound the PCT. Independent data are used to test the method.
6. Conclusions and Recommendations

6.1. Conclusions

The purpose of this report is to review and summarise the AEA's achievements in the area of LOCA uncertainty analysis since 1986 and to make recommendations for the next steps in this work.

We have therefore described the AEA methodology for evaluating the uncertainty in calculations of PWR LOCAs and two studies to demonstrate it. Because of various difficulties in modelling LOCA the uncertainty analysis is also difficult and the methodology needs to be justified appropriately from first principles. The starting point is that reliable advice is needed by people who have to take decisions about the design, siting, licencing and safe operation of plant.

The "AEA method" for uncertainty analysis consists of three elements:
- definition of a "reasonable uncertainty range"
- method of combining uncertainties and
- a flow chart for organising full scale plant analysis.

Parts of the method have been demonstrated in two sets of calculations, for the LOBI test BL-02 and for a PWR large LOCA. For these calculations both TRAC-PF1/MOD1 (for the LOBI calculation) and TRAC-PF1/MOD2 (for the PWR) were modified to allow modelling parameters to be introduced to physical subroutines from input.

For the LOBI test 20 calculations with TRAC-PF1/MOD1 were successfully run and reported. The predicted uncertainty ranges mostly bounded the data. The large LOCA calculations used TRAC-PF1/MOD2 and had to be curtailed due to poor mass conservation. However 26 calculations of the first 50 s of the transient were accomplished.

The two studies reported have revealed no difficulties in operating the AEA method that cannot be overcome in practice and no problems that invalidate the method. The LOBI calculations show that uncertainty ranges for calculated quantities can be successfully calculated. The AEA method is therefore recommended for further use.

6.2. Recommendations

The next step in this work should be to improve the conservation of mass in TRAC-PF1/MOD2 so that the PWR uncertainty study can be completed. The problem has been reported to the code authors at Los Alamos and to the USNRC for action through the Cose and Maintenance Program (CAMP). A main aim of CAMP is to ensure that users' important problems with codes covered by the agreement are resolved satisfactorily.
Additional uncertainties that have been identified as important should also be included. These are:

- Reflood modelling using the TRAC-PF1/MOD2 model
- Interphase heat transfer
- Moderator feedback
- Correlation with primary pressure peaks
- Steam generator tube rupture.

Because of the diversity of approaches to LOCA uncertainty analysis, international efforts to compare them and identify the most effective methods should continue. To this end the OECD NEA CSNI are considering proposals for an International Standard Problem and a Workshop to compare methods. These initiatives should be supported.
7. References


<table>
<thead>
<tr>
<th>Single Variation</th>
<th>Case</th>
<th>Parameter Variation</th>
<th>Blowdown PCT (K)</th>
<th>Post-Blowdown PCT (K)</th>
<th>Time of P-BPCT (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>HR1  HR2</td>
<td>HR1  HR2</td>
<td>HR1  HR2</td>
</tr>
<tr>
<td>Base case</td>
<td>BC2</td>
<td>(k_{\text{fuel}} + 0.8) (h_{\text{gen}} = 4200) (C_d \times 1.5)</td>
<td>984  1148</td>
<td>898  1066</td>
<td>21   23</td>
</tr>
<tr>
<td></td>
<td>F2</td>
<td>(k_{\text{fuel}} - 0.8) (h_{\text{gen}} = 240000)</td>
<td>758  947</td>
<td>726  919</td>
<td>25   26</td>
</tr>
<tr>
<td>Reactor Life - Fuel State</td>
<td>F1</td>
<td>(k_{\text{fuel}} + 0.0) (h_{\text{gen}} = 20000)</td>
<td>840  964</td>
<td>812  956</td>
<td>49   48</td>
</tr>
<tr>
<td></td>
<td>C1</td>
<td>(k_{\text{clad}} = -2.02)</td>
<td>992  1176</td>
<td>900  1064</td>
<td>22   22</td>
</tr>
<tr>
<td></td>
<td>C2</td>
<td>(k_{\text{clad}} = +2.02)</td>
<td>990  1165</td>
<td>893  1060</td>
<td>21   22</td>
</tr>
<tr>
<td></td>
<td>M1</td>
<td>Low</td>
<td>999  1189</td>
<td>886  1047</td>
<td>27   24</td>
</tr>
<tr>
<td></td>
<td>M2</td>
<td>High</td>
<td>1007  1167</td>
<td>954  1118</td>
<td>19   20</td>
</tr>
<tr>
<td></td>
<td>S1</td>
<td>(C_d \times 0.75) (L_q \times 1.4)</td>
<td>977  1171</td>
<td>896  1076</td>
<td>23   22</td>
</tr>
<tr>
<td></td>
<td>S2</td>
<td>(C_d \times 3.5) (L_q \times 0.6)</td>
<td>1011  1205</td>
<td>891  1067</td>
<td>20   22</td>
</tr>
<tr>
<td>Bubbly/Slug Interphase Drag</td>
<td>ED1</td>
<td>(D_d \times 0.6)</td>
<td>989  1186</td>
<td>899  1072</td>
<td>26   26</td>
</tr>
<tr>
<td></td>
<td>ED2</td>
<td>(D_d \times 1.4)</td>
<td>986  1183</td>
<td>893  1073</td>
<td>38   20</td>
</tr>
<tr>
<td></td>
<td>E1</td>
<td>(E \times 0.9)</td>
<td>990  1181</td>
<td>911  1090</td>
<td>36   36</td>
</tr>
<tr>
<td></td>
<td>E2</td>
<td>(E \times 1.3)</td>
<td>977  1169</td>
<td>897  1067</td>
<td>22   22</td>
</tr>
<tr>
<td></td>
<td>ST</td>
<td>(f_i \times 7.0)</td>
<td>996  1182</td>
<td>894  1070</td>
<td>21   21</td>
</tr>
<tr>
<td></td>
<td>FC</td>
<td>(h_{\text{DB}} \times 1.7)</td>
<td>967  1126</td>
<td>884  1038</td>
<td>21   23</td>
</tr>
<tr>
<td></td>
<td>H1</td>
<td>(\varepsilon - 0.08) (h_{\text{BB}} \times 0.8)</td>
<td>990  1169</td>
<td>943  1132</td>
<td>45   45</td>
</tr>
<tr>
<td></td>
<td>H2</td>
<td>(h_{\text{BB}} \times 2.0)</td>
<td>982  1163</td>
<td>795  921</td>
<td>22   22</td>
</tr>
<tr>
<td></td>
<td>H3</td>
<td>(\varepsilon + 0.20) (h_{\text{BB}} \times 4.0)</td>
<td>978  1172</td>
<td>730  831</td>
<td>22   23</td>
</tr>
<tr>
<td></td>
<td>T1</td>
<td>(T_{\text{min}} + 75.0)</td>
<td>1013  1175</td>
<td>926  1089</td>
<td>22   44</td>
</tr>
<tr>
<td></td>
<td>T2</td>
<td>(T_{\text{min}} + 150.0)</td>
<td>1012  1198</td>
<td>960  1148</td>
<td>33   45</td>
</tr>
</tbody>
</table>

This table is reproduced from Reference 6 where the notation and units are explained.

AEA RS 5522 - 19 -
<table>
<thead>
<tr>
<th>Case</th>
<th>Blowdown PCT (K)</th>
<th>Post-Blowdown PCT (K)</th>
<th>Time of Post-Blowdown PCT (s)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>HR1</td>
<td>HR2</td>
<td>HR1</td>
</tr>
<tr>
<td>1a</td>
<td>1042</td>
<td>1221</td>
<td>1016</td>
</tr>
<tr>
<td>1b</td>
<td>1017</td>
<td>1190</td>
<td>957</td>
</tr>
<tr>
<td>1c</td>
<td>997</td>
<td>1193</td>
<td>948</td>
</tr>
<tr>
<td>2a</td>
<td>738</td>
<td>917</td>
<td>571</td>
</tr>
<tr>
<td>2b</td>
<td>737</td>
<td>909</td>
<td>556</td>
</tr>
<tr>
<td>2c</td>
<td>766</td>
<td>938</td>
<td>548</td>
</tr>
</tbody>
</table>

This table is reproduced from Reference 6 where the notation is explained.
Figure 1: Flow Chart for Uncertainty Prediction for Reactor. From Reference 1.
Figure 2: LOBi BL-02 Experiment. Rig Data and Calculated Range, Total Primary System Mass. From Reference 1.
The following are plotted against reactor time.

**Max Hot Rod Temp**

<table>
<thead>
<tr>
<th>Symbol</th>
<th>Name</th>
<th>Units</th>
</tr>
</thead>
<tbody>
<tr>
<td>BC2</td>
<td>Max Hot Rod Temp</td>
<td>Deg. K</td>
</tr>
<tr>
<td>IA</td>
<td>Max Hot Rod Temp</td>
<td>Deg. K</td>
</tr>
<tr>
<td>2A</td>
<td>Max Hot Rod Temp</td>
<td>Deg. K</td>
</tr>
<tr>
<td>2C</td>
<td>Max Hot Rod Temp</td>
<td>Deg. K</td>
</tr>
</tbody>
</table>

Figure 3: 4 Loop Westinghouse PWR. Max Temp of Hot Rod 2. Base Case (BC2) and Upper and Lower PCTs (1a, 2b, 2c). From Reference 6.
Development and Application of an Uncertainty Methodology for the Sizewell B Large LOCA Safety Case

by

P Lightfoot¹ and M Trow²

1.0 INTRODUCTION

Sizewell B Power Station is based upon a four-loop Westinghouse Pressurised Water Reactor (PWR) with a core thermal rating of 3411 MW. It is the first commercial PWR to be constructed in the UK, and is currently in an advanced stage of commissioning. As part of the process of granting formal consent for the construction of Sizewell B, the government established a Public Inquiry which ran from 1983 to 1985. As a result of the Inquiry, concerns were raised with regard to the use of the evaluation methodology for LOCA analysis (as prescribed in 10CFR50 Appendix K). These concerns were related to the difficulties of quantifying the margins to safety, and hence in demonstrating that the method resulted in a conservative assessment of the Peak Cladding Temperature (PCT). The licensing framework in the UK, unlike the USA, is not prescriptive, and Nuclear Electric (formerly the Central Electricity Generating Board) decided in 1985 to adopt a LOCA licensing methodology based on the application of physically based mechanistic codes. It was understood from the outset that, in order to produce such a safety case for the Pre-Operational Safety Report (POSR), a methodology would be required for the treatment of uncertainties. NNC were contracted to develop an uncertainties methodology and to provide the large LOCA analysis.

Following the decision to adopt a mechanistic approach, a code review was conducted during 1986 and 1987 which culminated in the selection of the proprietary Westinghouse code WCOBRA/TRAC as the main thermal-hydraulic licensing code for large LOCA. The period of 1987-1991 involved a substantial programme of code development, verification, validation and nodalisation studies culminating in the production of a frozen code version for the formal safety analysis. Finally, during the period 1991-1992, the plant analysis was undertaken including the application of an uncertainties methodology which had been developed between 1986 and 1991. This work was supported by a substantial R&D programme. In addition, Nuclear Electric undertook a programme of independent assessment of the NNC analysis using the TRAC-PFI/MOD1 code. The Sizewell B large LOCA safety case has been formally submitted to the Nuclear Installations Inspectorate (NII).

2.0 OVERVIEW OF METHODOLOGY

Throughout the development of the methodology, a key requirement was that it could be applied within the available timescales and resource levels. A brief outline of the method is presented below. It will be noted that the method bears many similarities to the Code Scaling, Applicability and Uncertainty (CSAU) study undertaken by the USNRC. However, as outlined below, the method differs significantly in two specific areas.

- Select Code and Version - As outlined above, a code review was undertaken to assess the suitability of various code options. The selected code, WCOBRA/TRAC, was judged to

¹PWR Project Group, Nuclear Electric plc, Booths Hall, Chelford Road, Knutsford, Cheshire, WA16 8QG
²NNC Ltd, Booths Hall, Chelford Road, Knutsford, Cheshire, WA16 8QZ
satisfy all the necessary criteria and had the advantage of a very flexible noding capability for the vessel. Following a period of code development, a frozen code version was established for the formal validation and subsequent plant analysis.

- **Establishment of Validation Matrix** - In parallel with the code development, a dataset review was undertaken to evaluate the available experimental datasets for the major phenomena judged to be associated with large LOCA (e.g. downcomer bypass, critical flow, post-dryout heat transfer). This review was carried out as part of the supporting R&D programme by a team of independent UK experts, who classified the data in terms of factors such as quality, scale and relevance to plant conditions. Based on these recommendations, a validation matrix was constructed which addressed the majority of the important phenomena. The preference was to utilise full scale facility data wherever this was available, so as to minimise any uncertainties associated with scale.

- **Code Validation and Plant Nodalisation** - WCOBRA/TRAC was validated using the separate effects and integral effects tests identified in the validation matrix. Throughout the validation analysis, strict nodalisation and modelling guidelines have been adopted to ensure consistency within the various validation analyses. This minimises any effects due to nodalisation preferences of individual analysts, and the essential features of the validation nodalisations are transferred to the plant model.

- **Phenomena Ranking** - As the four-loop PWR considered in the CSAU analysis is similar to the Sizewell B PWR, the CSAU Phenomena Identification and Ranking Table (PIRT) was taken as the starting point for ranking the phenomena. This table was reviewed and modified by a team of NNC experts, based on the results of the validation calculations and experience gained from plant analysis.

- **Define Plant Operating Parameters** - The plant operating parameters were set at their bounding limits for full power operation. This approach has been adopted to ensure that the safety case is applicable at all times of life. It is different from the CSAU approach and has a significant effect on the final calculated PCT.

- **Define Accident Boundary Conditions** - The accident boundary conditions including break location, safeguards system operation and treatment of off-site power have all been set at conservative values. Where necessary, this has been achieved by undertaking specific sensitivity analysis (e.g. to define the worst break size and location).

- **Perform Plant Sensitivity Calculations** - Uncertainties in the highly ranked phenomena were addressed in the uncertainty analysis by performing plant sensitivity analyses. These calculations determined the sensitivity of the PCT to the phenomena by incorporating multipliers which modified specific models. The values of the multipliers were generally determined from the validation studies by comparing code predictions with experimental data, and including an additional allowance for experimental error. The ranges were judged to provide a reasonable bound on the data uncertainty. This approach differs from that of CSAU in that no specific level of confidence was attached to these multipliers, and it has the advantage that there is no need to assume a particular form for the probability distribution function. Plant sensitivity calculations were also undertaken to assess the uncertainties highlighted in the nodalisation development.

In addition to quantifying the sensitivity of the PCT to the phenomena and model uncertainties, sensitivity studies were also undertaken to demonstrate the absence of 'cliff-edge' effects. In such cases, the range of multipliers was wider than that which would be required to provide a reasonable bound to the experimental data.
Determine Effect of Combined Uncertainties - For the uncertainties which have the largest effect on the PCT, a number of combined uncertainty analyses were undertaken to determine whether the phenomena interact in a non-linear manner. A final combined uncertainty calculation was then undertaken, with the model uncertainties selected using a judgemental process based on the results of the previous single and combined uncertainty analyses. For the final combined uncertainty analysis, the multipliers were in some cases relaxed to provide a less extreme and more physical representation of the phenomena. A number of biases were then added to this PCT in order to obtain the overall PCT for the large LOCA safety assessment. These biases were separately evaluated and, although they may not be totally independent of the other uncertainties, it was judged conservative to add these biases as additional penalties.

The major difference between the Sizewell B method and CSAU is that the Sizewell B method does not attempt to provide a specified level of confidence in the PCT or the uncertainty ranges. Instead, engineering judgement is used to ensure that the final PCT has an associated high level of confidence. This is achieved by using bounding and conservative values in the assessment.

3.0 DETAILED APPLICATION OF METHOD

3.1 Uncertainties in Models and Phenomena arising from Code Validation

The overall conclusion of the validation study was that WCOBRA/TRAC is well validated for performing large LOCA plant analysis and that it adequately models the majority of the phenomena. However, for some key phenomena, there was sufficient uncertainty in the code to data comparison, the experimental uncertainties or the range of conditions covered, that for these phenomena further consideration was required as part of the overall assessment of uncertainties. These phenomena were addressed by performing plant sensitivity calculations.

The dataset reviews examined the quality of the data and the accuracy of the instrumentation as well as determining the applicability of the data to the plant transient. Uncertainty in the measured experimental initial and boundary conditions was minimised by selecting good quality test data. The residual uncertainties in the experimental data were addressed via the plant sensitivity calculations, as the code model multipliers were defined to encompass both experimental and calculational uncertainties.

The uncertainties in the models and phenomena were separately ranked for each of the blowdown, refill and reflood phases of a large LOCA as indicated in Table 1. With the exception of models which were known to be conservative, the majority of those phenomena or models ranked as high were investigated by plant sensitivity calculations, as indicated in Table 1. Those high ranking models which were not demonstrably conservative and had not been the subject of further sensitivity analysis were assumed to be bounded by uncertainties in other related or interacting phenomena. For example, uncertainties in rod heat transfer during the nucleate boiling regime was judged to be bounded by the assessed uncertainties in the post-dryout heat transfer regime.

The nodalisation uncertainties were minimised by applying consistent nodalisations throughout the validation study and conserving the significant features of these nodalisations in the plant model. The plant nodalisation was further supported by sensitivity studies. However, two important aspects of nodalisation were not addressed by the validation study, namely the hot assembly core channel and the upper head and guide tubes. The hot assembly model was demonstrated to be very conservative by plant calculations, so no further nodalisation study was required. The upper head and guide tube nodalisation was considered in plant sensitivity calculations as part of the uncertainty assessment.
The effects of scale were minimised by selecting full scale data wherever this was available. Examples of such facilities include Marviken for critical flow, the Upper Plenum Test Facility (UPTF) for downcomer bypass, upper plenum de-entrainment, steam binding and cold leg phenomena, ACHILLES for accumulator nitrogen blowdown modelling and post-dryout heat transfer, and FLECHT-SEASET for post-dryout heat transfer. The LOFT facility, which has a volume scale of 1/48th of a four loop PWR, has a short vessel and non-prototypic downcomer and upper plenum. It was shown in the CSAU analysis that the effects of scale are largely confined to the refill and reflood phases, and LOFT data were therefore used to investigate the interaction of phenomena during the blowdown phase where the effects of scale are at a minimum.

3.2 Uncertainties in Reactor Operating Conditions

The uncertainties in the reactor operating conditions were separately ranked for each of the blowdown, refill and reflood phases of a large LOCA as indicated in Table 2. Those parameters ranked as medium or high were set either at the safety analysis bounding limit or at a demonstrably conservative value. This invokes additional conservatism as the individual parametric values are not all achievable at the same time in life.

Examples of the specific assumptions include:

- Total reactor power was modelled as 102% to allow for measurement uncertainties.
- The overall power peaking factor was bounded by applying the limiting $F_Q$ value of 2.32.
- A uniform chopped cosine was used to defined the axial power profile. Sensitivity calculations undertaken for the two extremes of allowable axial offset demonstrated the conservatism of this assumption.
- The radial power profile across the core was modelled by low, average and hot assembly regions. The hot rod was located within the hot assembly. WCOBRA/TRAC does not predict a discernible 'chimney' effect for the hot assembly thermal hydraulics, and the rods in the hot assembly and the hot rod therefore achieved significantly higher clad temperatures than the average rods.

3.3 Uncertainties in Accident Boundary Conditions

The uncertainties in the accident boundary conditions were also separately ranked for each of the blowdown, refill and reflood phases of a large LOCA as indicated in Table 3. Those parameters ranked as medium or high were set at conservative bounding values.

Examples of the specific boundary conditions applied include:

- Decay heat was modelled using the ANS 79 standard plus uncertainties, with input conditions selected to provide values of decay heat which are bounding at all times of life.
- Control rods were assumed to fail to insert on reactor trip.
- The base case scenario was a double-ended cold leg guillotine break with a break flow multiplier of unity, which sensitivity analysis showed to be the limiting case.
- Minimum Emergency Core Cooling System (ECCS) safeguards was modelled so as to maximise the PCT. Injection was modelled as being into the two intact cold legs closest to the broken loop so as to maximise the downcomer bypass during the blowdown phase.
• The containment pressure was set conservatively low, so as to maximise steam binding during reflood, by assuming maximum containment safeguards.

• Off-site power was modelled as being available, as sensitivity analysis showed this to be slightly more conservative than assuming loss of off-site power.

• The broken loop Reactor Coolant Pump (RCP) impeller was assumed to lock prior to reflood.

• Steam generator reverse heat transfer was maximised by isolating the steam and feedlines, so as to increase the effect of steam binding during reflood.

• The hot assembly was located so as to minimise its cooling during upper head blowdown.

3.4 Code Uncertainties

A number of uncertainties exist in any large thermal-hydraulics code due to approximations in the models and correlations. Such uncertainties were effectively addressed in the validation study, and did not need to be separately assessed.

The possibility of significant coding errors was judged to be negligible, due to the maturity of WCOBRA/TRAC, its extensive application, and the specific verification which was conducted on all the major calculational routines. Any remaining coding errors were likely to be of a minor nature, and were judged not to significantly affect the validation results.

3.5 Base Case and Sensitivity Analyses

As indicated in Table 1, the uncertainty in twelve phenomena or models was investigated by plant sensitivity calculations. The uncertainties were bounded by applying multipliers to specific models. The multipliers were determined by comparing the calculated results with the experimental data such that the code predictions bounded the data after allowance was made for experimental errors. Where no relevant data was available, large ranges were applied to the multipliers to assess the sensitivity of the PCT to the particular phenomena and to ensure there were no 'cliff-edge' effects.

The critical flow, post-dryout heat transfer, steam binding and hot leg slip are examples of where multipliers were based on code comparisons with experimental data. Examples of uncertainties for which no experimental data comparisons were utilised include pump degradation, core entrainment and upper head blowdown.

The individual uncertainty analyses addressed the limits of the uncertainty range both to determine the sensitivity of the PCT and to demonstrate that no 'cliff-edge' effects were present. In the majority of cases, the uncertainty ranges in the phenomena and models considered were extreme, and in some cases unphysical.

The nominal base case analysis simulated a double-ended cold leg guillotine break with a break flow multiplier of unity. The reactor and accident boundary conditions were set at demonstrably conservative or bounding values, and off-site power was assumed available. The only adjustment to any model was to apply a multiplier of 0.9 to the two-phase break flow model, as this was the value which gave the best fit to the Marviken data.

The PCT for the base case analysis was 1202K. Table 4 lists the sensitivity calculations and the PCT difference from the base case analysis. A number of calculations were not continued beyond
the start of reflood due, in general, to the reflood dynamics uncertainty which is described below. For these cases, the PCT temperature increase over the base case is quoted at the start of reflood.

3.6 Reflood Dynamics

The early stages of reflood are affected by a number of interacting phenomena. However, the main uncertainty is in the timing of the accumulator nitrogen blowdown and the interaction of the resultant pressurisation with the oscillatory reflood flow. The complexity of the interacting phenomena occurring during the accumulator nitrogen blowdown phase (termed reflood dynamics) is illustrated by the following description.

The pressurisation of the intact cold legs and the top of the downcomer region forces liquid from these regions out through the break and into the containment. The pressurisation also lowers the downcomer liquid level forcing a large plug of liquid up through the core. The amount of liquid forced through the core at this time is dependent on the pressurisation of the top of the downcomer, which is influenced by the venting capacity of the break. The water movement in this region prior to nitrogen blowdown is governed by condensation effects in the intact cold legs and downcomer. The water which passes through the core partly de-entains in the upper plenum and partly flows into the steam generator tubes, via the hot legs, where it vaporises, resulting in steam binding. The liquid which has de-entrained in the upper plenum can drain back into the core. This liquid will vaporise, improving the rod to fluid heat transfer but retarding the bottom-up quench front progression. The low core liquid level and the consequently higher downcomer liquid level produced as a result of the increased steam generation in the core can significantly increase the vessel side break flow.

To investigate the uncertainty to the reflood dynamics effect, the plant uncertainty calculations were examined to determine the variations in the time difference between the start of reflood and initiation of the nitrogen blowdown phase. The time window was found to be \(-5.2\) to \(+2.6\) seconds with respect to the base case analysis. This time window also takes account of uncertainties in the initial accumulator water volume and nitrogen pressure. A total of eleven sensitivity calculations were performed to determine the effect of the sensitivity on the reflood behaviour and on the PCT. The results showed an apparently random effect on the PCT with the maximum increase over the base case being \(+44K\), and with no calculation falling below the base case PCT value. This bias was increased to 50K to allow for possible small increases due to the time variations which were not specifically assessed. As this uncertainty was essentially random and solely due to a timing effect it was treated as a bias, with the full penalty being conservatively added to determine the overall PCT.

The variation of up to 50K on the PCT associated with the reflood dynamics uncertainty calculation made it difficult to quantify the individual parameter uncertainties during reflood. For this reason, many individual uncertainty calculations were not continued beyond the start of reflood.

3.7 Uncertainty Interaction Calculations

It can be seen from Table 4 that the largest uncertainties in the PCT arose from the critical flow and post-dryout heat transfer phenomena. To examine possible interactions of these uncertainties, additional plant analyses were undertaken to assess each of the four possible combinations of the minimum and maximum values for the two phenomena. The results of these analyses showed that the phenomena combine in an almost linear manner. The results also confirmed that the worst combination was nominal critical flow and minimum post-dryout heat transfer.
3.8 Combined Uncertainties Calculation

A combined uncertainties calculation was used to calculate the final PCT, before the addition of biases. Each individual uncertainty calculation was examined to determine the significance of the phenomena in terms of the deviation from the base case analysis. Criteria used in the assessment were the effect on the overall calculation and the change to the PCT. Where a physical phenomena was judged to have a potentially significant impact, that phenomena was included in the overall uncertainty calculation.

The phenomena included were:

- **Off-site power** - assumed available as this produced a PCT penalty compared with the case where off-site power was not available.

- **Critical flow** - set to the nominal value as this produced the highest PCT compared to the upper and lower extremities of the uncertainty range.

- **Post-dryout heat transfer** - a multiplier of 0.85 was used in the combined uncertainties calculation for both blowdown and reflood. The 0.7 reflood factor used in the individual uncertainty analysis was judged to be overly conservative, being primarily used to investigate the existence of 'cliff-edge' effects. The factor of 0.85 applied in reflood still provided a conservative bound of the radially averaged PCTs from the experimental data.

- **Pump two-phase behaviour** - had a significant influence on the blowdown PCT, producing a penalty of 31K at the start of reflood. The full two standard deviations in the head and torque multipliers were retained in the overall combined uncertainty calculation.

- **Slip in intact cold legs** - set to unity as it significantly influenced the PCT during blowdown and produced a penalty of 27K at the start of reflood.

- **Condensation in intact cold legs** - was reduced by a factor of 10 from the nominal value. The original uncertainty, which totally suppressed condensation, was judged to be too extreme for the combined uncertainty calculation.

- **Steam binding** - the delayed calculation of steam binding was included in the combined uncertainty calculation as it gave a better, though still conservative, estimate of this phenomenon. The sensitivity calculation gave a PCT benefit of 39K.

The phenomena not included in the combined uncertainties calculation were:

- **Minimum film boiling temperature** - was used in the code to predict both quench front progression and spontaneous rewet. The validation showed that quench front progression during blowdown was very conservatively predicted, whilst a reasonable prediction was obtained during reflood. The lower limit of the spontaneous rewet temperature was not bounded at high pressures by the minimum film boiling temperature model. However, for large LOCA analysis, it was judged to be the quench front progression which provided the dominant influence on PCT rather than spontaneous rewet. It was judged that the conservatism in the high pressure quench front progression plus the application of post-dryout heat transfer sensitivity multipliers was adequate to bound any uncertainties in the minimum film boiling temperature.

- **Accumulator nitrogen blowdown** - was not included as uncertainties in this model did not produce a PCT penalty.
Upper head blowdown - was not included as both uncertainty calculations showed a benefit in PCT at the start of reflood.

Hot leg slip - was not included as uncertainties did not have a significant effect on the PCT.

Condensation ramp - was not included as its uncertainty is in conflict with the suppression of condensation, which showed a larger PCT penalty.

Core entrainment - was not included as its uncertainty did not significantly affect the PCT.

Table 5 presents the temperature differences between the combined uncertainties calculation and the base case, together with the equivalent results for the single uncertainty calculations. The PCT differences are separately quoted for the start of reflood and for the overall PCT.

For the start of reflood, the penalty associated with the heat transfer uncertainty was partly due to a delay in the start of reflood and partly due to the reduced heat transfer. The penalties associated with the pump degradation, cold leg slip and cold leg condensation uncertainty calculations were all caused by a delay to the start of reflood. The combined uncertainty calculation showed the individual uncertainties were not additive, and a substantial delay to the start of reflood was not predicted.

A comparison for the final PCT was more difficult due to the random influence of the reflood dynamics on the calculations, which produced a variation of up to 50K in the predicted PCTs. An examination of the individual analyses indicated that the base case, pump degradation, steam binding and combined uncertainties calculations all produced favourable refloods, whereas the heat transfer and cold leg condensation analyses produced unfavourable refloods. If only a single calculation is considered, it is impossible to separate out the effect of the uncertainty due to reflood dynamics from the effect of the individual uncertainty under consideration. However, it can be argued that the reduced heat transfer calculation produced a PCT penalty whereas the simulated delay to the calculation of steam binding produced a benefit.

If we consider that the pump degradation, cold leg slip and cold leg condensation uncertainties produced a random effect during reflood, then the combined uncertainties result is seen to be consistent with the individual uncertainties. If a less favourable reflood had resulted for the combined uncertainties calculation, then a PCT penalty of +76K might have resulted. PCT penalties in the range +26K to +76K appear to be consistent with the individual uncertainties given the competing effects of the heat transfer and steam binding uncertainties, the reduced ranges for post-dryout heat transfer and cold leg condensation uncertainties, and the random variation produced by the reflood dynamics.

3.9 Overall Bounding Peak Clad Temperature

The overall bounding value for the PCT was determined from the uncertainty analysis by taking the results of the combined uncertainties calculation and adding three bounding biases.

The biases included were:

- Reflood dynamics - has been extensively discussed. It was present, to some degree, in every calculation, and the simplest way to treat it was to take the maximum value and apply it as a bias.

- Clad ballooning - was assessed using the Westinghouse code BART-A1. BART-A1 is a dynamic ballooning code which has been extensively validated against a variety of separate
effects and clad ballooning experiments. For the plant analysis, the necessary boundary conditions were supplied by WCOBRA/TRAC. A variety of calculations were undertaken to assess the sensitivity to various parameters, and to ensure that no 'cliff-edge' effects existed. The final bounding bias was obtained by imposing a very conservative limiting blockage profile derived from a review of dynamic clad ballooning experiments. This large conservatism allows for uncertainties in the WCOBRA/TRAC boundary conditions and enables the clad ballooning penalty to be treated as a bias.

- **Pressuriser location** - a sensitivity calculation, undertaken once the full uncertainty analysis was underway, revealed a small PCT penalty prior to reflood if the pressuriser was located in the broken loop. This was not considered to be significant in terms of the overall PCT, but was nevertheless included as a conservative bias.

The overall PCT was obtained by summating the combined uncertainty penalty of 1228K, together with the biases for reflood dynamics (+50K), clad ballooning (+110K) and pressuriser location (+12K), giving a final bounding PCT of 1400K. This gave a margin of 77K to the fuel rod structural integrity limit of 1477K.

### 4.0 BETTER ESTIMATE CALCULATION

The base case analysis used for the uncertainty analysis included a large number of conservatism. A better estimate large break LOCA analysis was therefore undertaken to demonstrate that the PCT could be significantly reduced by the use of more realistic plant operating parameters and boundary conditions.

The relaxations applied for the better estimate calculation were:

- **Core power** - was set at its nominal 100% value.
- **Fuel stored energy** - was modelled using bounding values representative of base load operation.
- **Decay heat** - was modelled for a typical equilibrium cycle.
- **Peaking factors** - were modelled for a typical equilibrium cycle.
- **Accumulators** - were all assumed to be operational.
- **Safety injection system** - pumps were all assumed to be operational.
- **Steam generators** - were all modelled with feedwater flow maintained.

The better estimate calculation produced a PCT of 872K, representing a reduction of 330K on the PCT of 1202K given by the base case calculation described in 3.5 above.

### 5.0 RESOURCE REQUIREMENTS

The methodology required 13 sensitivity calculations to determine the boundary conditions in terms of break location, break size, etc.. The uncertainty analysis itself comprised 13 calculations simulating the whole transient, with a further 24 modelling only part of the transient. In addition, 18 BART-A1 calculations were undertaken to investigate the clad ballooning phenomenon.
If the methodology were to be applied to a nuclear plant of similar design to Sizewell B, it is judged that the maximum number of calculations which would be required are 13 full analyses and 13 part-transient analyses. Only three BART-A1 clad ballooning analyses would be required.

6.0 SUMMARY AND CONCLUSIONS

This paper presents an uncertainty methodology which has been successfully applied to the licensing of Sizewell B for large break LOCA. The emphasis of this approach has been on gaining a detailed understanding of the physical processes and of the sensitivity to individual phenomena. The major contributors to uncertainty have been identified, and have subsequently been included in a combined uncertainty analysis.

The combined uncertainty analysis demonstrated that uncertainties did not combine in a highly non-linear manner. Phenomena such as the random reflood effect and clad ballooning have been treated as bounding biases in the assessment of the overall bounding peak clad temperature.

The plant initial and boundary conditions have been conservatively defined for the uncertainty analysis. A better estimate calculation, which uses more realistic assumptions, shows a large benefit in the predicted peak clad temperature, thereby demonstrating the conservatism of the uncertainty analysis.

The UK licensing regime is not prescriptive in terms of the approach to large LOCA analysis, and no attempt has been made to apply a formal probability or confidence limit to the final PCT. Rather, reliance has been placed on engineering judgement, to ensure that the final bounding peak clad temperature is conservative.

The Sizewell B uncertainty analysis was completed within the timescale and resources limitations. It has been shown to be practical in its application and reductions in the required analysis scope have been identified for any future plants of similar design.

7.0 ACKNOWLEDGEMENT

This Paper describes the work of a number of NNC engineers performed on behalf of, and with the collaboration of, the PWR Project Group of Nuclear Electric plc.
<table>
<thead>
<tr>
<th>Phenomenon / Model</th>
<th>Rank</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Blowdown</td>
</tr>
<tr>
<td>Fuel rod material properties</td>
<td></td>
</tr>
<tr>
<td>- thermal conductivity</td>
<td>L</td>
</tr>
<tr>
<td>- specific heat</td>
<td>L</td>
</tr>
<tr>
<td>- gap conductance</td>
<td>H</td>
</tr>
<tr>
<td>Rod heat transfer</td>
<td>H</td>
</tr>
<tr>
<td>Post-dryout heat transfer</td>
<td>*</td>
</tr>
<tr>
<td>Minimum film boiling temperature</td>
<td>*</td>
</tr>
<tr>
<td>Critical heat flux</td>
<td>M</td>
</tr>
<tr>
<td>Entrainment in core</td>
<td>*</td>
</tr>
<tr>
<td>Interfacial drag in core</td>
<td>M</td>
</tr>
<tr>
<td>Interfacial heat transfer in core</td>
<td>*</td>
</tr>
<tr>
<td>Grid model</td>
<td>+</td>
</tr>
<tr>
<td>Hot assembly model</td>
<td>+</td>
</tr>
<tr>
<td>Critical flow at break</td>
<td>*</td>
</tr>
<tr>
<td>Critical flow in surge line</td>
<td>M</td>
</tr>
<tr>
<td>Pump two-phase behaviour</td>
<td>*</td>
</tr>
<tr>
<td>Hot leg slip in blowdown</td>
<td>*</td>
</tr>
<tr>
<td>Upper head blowdown</td>
<td>*</td>
</tr>
<tr>
<td>Fission heat model</td>
<td>+</td>
</tr>
<tr>
<td>Flashing in pressuriser</td>
<td>M</td>
</tr>
<tr>
<td>Lower plenum sweep-out</td>
<td>M</td>
</tr>
<tr>
<td>Cold leg phenomena</td>
<td>*</td>
</tr>
<tr>
<td>ECCS downcomer bypass</td>
<td>+</td>
</tr>
<tr>
<td>Downcomer wall heat transfer</td>
<td>M</td>
</tr>
<tr>
<td>CCFL at nozzle and core plates</td>
<td>L</td>
</tr>
<tr>
<td>Steam binding during reflood</td>
<td>*</td>
</tr>
<tr>
<td>Accumulator nitrogen model</td>
<td>*</td>
</tr>
<tr>
<td>Dissolved nitrogen</td>
<td>-</td>
</tr>
<tr>
<td>Accumulator line losses</td>
<td>-</td>
</tr>
<tr>
<td>Phenomenon / Model</td>
<td>Blowdown</td>
</tr>
<tr>
<td>--------------------------------------------------------</td>
<td>----------</td>
</tr>
<tr>
<td>Condensation rate</td>
<td>*</td>
</tr>
<tr>
<td>Metal / water reaction</td>
<td></td>
</tr>
<tr>
<td>Entrainment at top of downcomer in reflood</td>
<td></td>
</tr>
<tr>
<td>Break flow during reflood</td>
<td></td>
</tr>
<tr>
<td>Lower plenum wall heat transfer</td>
<td></td>
</tr>
<tr>
<td>Phenomena interaction</td>
<td></td>
</tr>
<tr>
<td>Reflood dynamics</td>
<td>*</td>
</tr>
</tbody>
</table>

**Key**

+ Placed at conservative bound

* Sensitivity analysis undertaken

<table>
<thead>
<tr>
<th>Rank</th>
</tr>
</thead>
<tbody>
<tr>
<td>H</td>
</tr>
<tr>
<td>M</td>
</tr>
<tr>
<td>L</td>
</tr>
</tbody>
</table>
### Table 2: Uncertainties in Plant Operating Conditions

<table>
<thead>
<tr>
<th>Plant Operating Condition</th>
<th>Rank</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Blowdown</td>
</tr>
<tr>
<td>Total reactor power</td>
<td>H</td>
</tr>
<tr>
<td>Power peaking factors</td>
<td>H</td>
</tr>
<tr>
<td>Axial power profile</td>
<td>M</td>
</tr>
<tr>
<td>Radial power profile</td>
<td>H</td>
</tr>
<tr>
<td>Fuel internal pressure</td>
<td>L</td>
</tr>
<tr>
<td>Fuel stored energy / Initial fuel temperatures</td>
<td>H</td>
</tr>
<tr>
<td>Fuel burnup</td>
<td>H</td>
</tr>
<tr>
<td>Primary circuit pressure</td>
<td>L</td>
</tr>
<tr>
<td>Secondary circuit pressure</td>
<td>L</td>
</tr>
<tr>
<td>Coolant temperature</td>
<td>L</td>
</tr>
<tr>
<td>Initial loop flows</td>
<td>L</td>
</tr>
</tbody>
</table>

**Key**

- **H** High
- **M** Medium
- **L** Low
<table>
<thead>
<tr>
<th>Accident Boundary Conditions and Assumptions</th>
<th>Blowdown</th>
<th>Refill</th>
<th>Reflood</th>
</tr>
</thead>
<tbody>
<tr>
<td>Decay heat</td>
<td>L</td>
<td>H</td>
<td>H</td>
</tr>
<tr>
<td>Control rod insertion</td>
<td>M</td>
<td>L</td>
<td>L</td>
</tr>
<tr>
<td>Break size</td>
<td>H</td>
<td>H</td>
<td>H</td>
</tr>
<tr>
<td>Break location</td>
<td>H</td>
<td>H</td>
<td>H</td>
</tr>
<tr>
<td>Break opening time</td>
<td>L</td>
<td>L</td>
<td>L</td>
</tr>
<tr>
<td>Injection flow characteristics</td>
<td>L</td>
<td>M</td>
<td>M</td>
</tr>
<tr>
<td>Injection water temperature</td>
<td>L</td>
<td>M</td>
<td>M</td>
</tr>
<tr>
<td>Accumulator water volume</td>
<td>L</td>
<td>M</td>
<td>M</td>
</tr>
<tr>
<td>Accumulator water temperature</td>
<td>L</td>
<td>M</td>
<td>M</td>
</tr>
<tr>
<td>Accumulator nitrogen pressure</td>
<td>L</td>
<td>M</td>
<td>M</td>
</tr>
<tr>
<td>Pressuriser water volume</td>
<td>L</td>
<td>L</td>
<td>L</td>
</tr>
<tr>
<td>Pressuriser water temperature</td>
<td>L</td>
<td>L</td>
<td>L</td>
</tr>
<tr>
<td>Containment backing pressure</td>
<td>L</td>
<td>L</td>
<td>H</td>
</tr>
<tr>
<td>Availability of off-site power</td>
<td>H</td>
<td>H</td>
<td>H</td>
</tr>
<tr>
<td>Locking of RCP impellers</td>
<td>L</td>
<td>L</td>
<td>H</td>
</tr>
<tr>
<td>Steam generator reverse heat transfer</td>
<td>L</td>
<td>L</td>
<td>H</td>
</tr>
<tr>
<td>Location of hot assembly</td>
<td>H</td>
<td>L</td>
<td>L</td>
</tr>
<tr>
<td>Location of operational accumulators and injection trains</td>
<td>H</td>
<td>L</td>
<td>L</td>
</tr>
</tbody>
</table>

**Key**

- **H** High
- **M** Medium
- **L** Low

**Note:** Coincident steam generator tube rupture is outside of the design basis.
Table 4: Plant Base Case and Sensitivity Analyses

<table>
<thead>
<tr>
<th>Uncertainty</th>
<th>Parameter</th>
<th>Multiplier / Value</th>
<th>$\Delta T$ (K)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Base case</td>
<td>Break discharge rate</td>
<td>Nominal:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Subcooled 1.0</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Two-phase 0.9</td>
<td></td>
</tr>
<tr>
<td>Critical flow</td>
<td>Break discharge rate</td>
<td>Maximum:</td>
<td>-11</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Subcooled 1.37</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Two-phase 1.10</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Minimum:</td>
<td>-90</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Subcooled 0.81</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Two-phase 0.78</td>
<td></td>
</tr>
<tr>
<td>Post-dryout heat transfer</td>
<td>Heat transfer factor and interfacial</td>
<td>Maximum:</td>
<td>-33</td>
</tr>
<tr>
<td></td>
<td>heat transfer factor</td>
<td>Blowdown 1.0</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Reflood 1.3</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Minimum:</td>
<td>+92</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Blowdown 0.85</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Reflood 0.7</td>
<td></td>
</tr>
<tr>
<td>Pump degradation</td>
<td>Head and torque curves</td>
<td>Plus two standard deviations</td>
<td>+31</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(start of reflood)</td>
<td></td>
</tr>
<tr>
<td>Quenching behaviour</td>
<td>Minimum film boiling temperature</td>
<td>Correlation -50K at high pressure</td>
<td>+32</td>
</tr>
<tr>
<td>Core entrainment</td>
<td>Entrainment rate</td>
<td>Maximum:</td>
<td>+1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1.5</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Minimum:</td>
<td>+2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.5</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>(start of reflood)</td>
<td></td>
</tr>
<tr>
<td>Uncertainty</td>
<td>Parameter</td>
<td>Multiplier / Value</td>
<td>$\Delta T$ (K)</td>
</tr>
<tr>
<td>-------------------------------------------</td>
<td>------------------------------------------------</td>
<td>--------------------</td>
<td>------------------</td>
</tr>
<tr>
<td>Intact cold leg phenomena</td>
<td>Interphase slip</td>
<td>1.0</td>
<td>$+27$ (start of reflood)</td>
</tr>
<tr>
<td></td>
<td>Interphase heat transfer</td>
<td>0.0</td>
<td>$+50$</td>
</tr>
<tr>
<td>Accumulator nitrogen blowdown</td>
<td>Nitrogen pressure</td>
<td>50% reduction</td>
<td>Negligible</td>
</tr>
<tr>
<td></td>
<td></td>
<td>No blowdown</td>
<td>-82</td>
</tr>
<tr>
<td>Upper head blowdown</td>
<td>Guide tube loss coefficient</td>
<td>200%</td>
<td>$-18$ (start of reflood)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>20%</td>
<td>$-18$ (start of reflood)</td>
</tr>
<tr>
<td>Steam binding</td>
<td>Liquid carryover to steam generators</td>
<td>30 seconds delay</td>
<td>-39</td>
</tr>
<tr>
<td>Hot leg slip</td>
<td>Interphase slip distribution parameter ($C_0$)</td>
<td>1.0</td>
<td>$+4$ (start of reflood)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1.3</td>
<td>$+11$ (start of reflood)</td>
</tr>
<tr>
<td>Reflood</td>
<td>Initiation of nitrogen blowdown</td>
<td>-5.2s to +2.6s</td>
<td>$+44$ (maximum from 11 calculations)</td>
</tr>
<tr>
<td>Effect of non-condensibles on condensation</td>
<td>Condensation ramp</td>
<td>Unrestrained condensation</td>
<td>$+12$ (start of reflood)</td>
</tr>
</tbody>
</table>
### Table 5: Combined Uncertainties Results

<table>
<thead>
<tr>
<th>Uncertainty</th>
<th>Effect on Peak Clad Temperature (ΔT)</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Start of Reflood (K)</td>
<td>Final PCT (K)</td>
</tr>
<tr>
<td>Heat transfer</td>
<td>*</td>
<td>+29</td>
<td>+92</td>
</tr>
<tr>
<td>Pump degradation</td>
<td></td>
<td>+31</td>
<td>-32</td>
</tr>
<tr>
<td>Cold leg slip</td>
<td></td>
<td>+27</td>
<td>-32</td>
</tr>
<tr>
<td>Cold leg condensation</td>
<td>*</td>
<td>+16</td>
<td>+50</td>
</tr>
<tr>
<td>Steam binding</td>
<td></td>
<td>-</td>
<td>-39</td>
</tr>
<tr>
<td>Combined uncertainties</td>
<td></td>
<td>+40</td>
<td>+26</td>
</tr>
<tr>
<td>Combined uncertainties including reflood dynamics bias of 50K</td>
<td></td>
<td>+40</td>
<td>+76</td>
</tr>
</tbody>
</table>

**Key**
- * Less conservative model variation applied for combined uncertainties calculation.
SPECIAL WORKSHOP ON UNCERTAINTY ANALYSIS METHODS

GRS Method (Germany)

H. Glaeser, E. Hofer (GRS)

a) Why do we study uncertainties?

Best estimate thermal-hydraulic computer codes are used to calculate postulated accidents in a realistic and not in a conservative way. Computer codes can approximate the physical behaviour with more or less accuracy. During the usual code validation process agreement of calculated results with experimental data is often obtained by choosing specific code input options or by changing parameter values in the model equations. Even if a frozen code version is used the comprehensive range of uncertainty in the results from application to power plants cannot be readily derived from this validation process. When the best estimate codes are used to evaluate the safety margins of nuclear power plants, quantification of the uncertainty in their prediction is highly desirable.

Three main incentives exist to study uncertainties of two-phase flow computer code predictions:

Firstly, within the licensing procedures there is the desire to replace the conservative requirements for code application by a best estimate concept supplemented by an uncertainty analysis to account for predictive uncertainties of the codes.

Secondly, modern transient two-phase flow codes form an essential basis for the elaboration of accident management procedures to mitigate abnormal beyond design basis accident plant conditions. For this purpose codes must be utilised in a "best-estimate" manner and the uncertainty must be quantified.

Thirdly, an interest in performing uncertainty analysis is related to code development. Determining the major sources of uncertainty in the code can help guide its further development in a cost effective manner. That means, uncertainty and sensitivity analysis can be used to prioritise further code development or further experimental investigations to improve the knowledge about the respective uncertain parameters and models.
Sensitivity measures of their contribution to code output uncertainty are simultaneously derived during an uncertainty analysis applying the GRS Method. Code deficiencies or significant inaccuracies will imply larger uncertainty ranges for the corresponding parameters and models (based on a sufficient number of prior code validation calculations) and consequently for key output parameters than in the case of a good code model which addresses the special phenomenon under consideration adequately.

An uncertainty analysis is not mandatory at present for the licensing procedure in Germany. Best estimate calculations can be performed in combination with conservative initial and boundary conditions. Beyond the maximum fuel rod cladding temperature not to exceed 1200°C, the German licensing procedure requires to prove that cladding tube defects will not exceed 10%, for example. Therefore, an uncertainty statement for important time-dependent code results are necessary, not only the uncertainty of the peak cladding temperature. The GRS Method is capable of providing the uncertainty of selected time-dependent (continuous-valued) output parameters. CSAU and GE Methods address only the uncertainty of a single-valued parameter, like peak cladding temperature or minimum vessel liquid level.

b) Identification of uncertainties

The following sources contribute to the overall uncertainty of computer code calculation results/1/:  

1) Selection of the scenario: Is the scenario adequate to answer the particular reactor safety question?

2) Physical model: Are all relevant mechanisms, phenomena and processes identified and described in sufficient detail (e.g. counter current flow, condensation, quenching, etc.)?

3) Mathematical formulation of the physical model: Are the number of conservation equations, the constitutive equations, correlations and component models an adequate representation of the physical model? The fundamental equations use space averaged values over the size of the mesh cells instead of local values. One of the major sources of uncertainty in current thermal hydraulic computer code simulation is the fact that it is not adequately known for all flow situations, how to describe the local
interfacial area density and the momentum, mass, and energy transfer rates at those interfaces, irrespective of the computational cell size.

4) Selection of parameter values: Thermal hydraulic parameter values are often obtained by adjusting the calculation results to measured data of particular experiments. Are the parameter values adequate for the mathematical formulation of the physical model and appropriate to answer the safety question with respect to a specific plant?

5) Numerical approximation: What is the error from discretisations in space and time, iterative solution processes, rounding, numerical instability, etc.?

6) Programming: What is the impact of programming errors?

7) Code user: The code user sets up the input deck. Was erroneous input provided, a false model option selected, etc.?

8) Computing system: Differences in code results using different computer systems, operating systems or compilers?

c) Characterisation of uncertainties

**Accuracy** "is a measure, usually expressed statistically, of the difference between measured and predicted quantities taking into account uncertainties and biases in both" /2/. The evaluation of the correct simulation of all relevant phenomena as well as of the performance of a computer code is an objective of the code validation process. It is common practice that the characterisation of the difference between experiment and calculation is mostly based on engineering judgements (such as "good agreement") without defining quantitative accuracy criteria. Ambrosini et al. /3/ and Kmetík et al. /4/ made proposals to quantify accuracy.

**Bias** "is a measure, usually expressed statistically, of the systematic difference between a true mean value and a predicted or measured mean" /2/.

**Error** is defined as the difference between a true value and a measured or calculated value. In most situations, it is not possible to determine what the error exactly is since the true value is unknown. We can only estimate what the error might be, and express it by the limits that one feels bound the possible error. These limits describe an
"uncertainty interval". The main sources of errors in computer code calculations are listed under b).

Uncertainty "is a measure of the scatter in experimental or predicted data" /2/. The term "uncertainty" refers to "a possible value that an error may have" /5/. The term "uncertainty" is used referring to the interval around a measured or calculated value within which the true value is believed to lie. For calculated values this interval is often given as a (e.g. 95%) subjective probability interval. It is distinguished between probability expressing stochastic variability and "subjective" probability expressing imprecise knowledge.

The GRS uncertainty method /6/ can address uncertainty from sources one to five under b). This kind of uncertainty can be characterised as follows:
- There is a set of alternatives,
- it is unknown which of these is true,
- only one alternative can be true and it is believed by those performing the analysis that the true alternative (or at least a sufficiently adequate alternative) is among those in the set.

All uncertainties of this kind can be described as parameter uncertainties. Model uncertainties are transferred into parameter uncertainties by introducing new uncertain parameters like corrective adders and multipliers or indices of alternative model formulations. Parameter uncertainties are for example model uncertainties, uncertain initial and boundary conditions, material properties, nodalisation changes, numerical parameters, and for the power plant uncertain parameters describing the state and operating conditions as well as modelling uncertainties of scale effects.

In order to prevent errors of source 7 under b) (code user effects) a complete and correct code documentation and user guidelines are necessary. Validation helps the code user to acquire expertise in code application and to reduce code user effects. An extensive code verification and validation and portability testing is necessary to prevent errors of sources 6 and 8 under b). Numerical benchmarks are part of the verification. Code validation results are a basis to quantify uncertainties of sources 2, 3, 4, and 5.
d) Quantification of uncertainties

The important phenomena and the associated uncertain model parameters in the investigated computer code and for the calculation to be performed are to be identified. This step includes to identify the uncertain initial and boundary conditions of the experiment or reactor plant scenario to be calculated. For each of the parameters the range (minimum and maximum) of possibly applicable values is to be determined. Subjective probability distributions of each of the individual uncertain parameters are specified to quantitatively express the corresponding state of knowledge. This is to account for the fact that evidence from previous code validation (or experimental evidence) indicates that the appropriate parameter value is more likely to be found in certain subranges of the given range than in others. The probability distribution is called "subjective" since it expresses the state of knowledge of fixed parameter values rather than stochastic variability.

Code effects, including numerical approximation due to the noding scheme, have to be evaluated during the code validation process. Finding the optimal noding, to describe the important phenomena, is a task of code validation. However, alternative noding schemes can be included in the uncertainty analysis.

All potentially important uncertain parameters and model formulations which are identified by the experts will be treated by the analysis. There is no limitation to a small number of uncertainties since the number of code calculations does not grow with the number of uncertain parameters in the GRS Method /1/,/6/. State of knowledge dependence between parameters can be taken into account quantitatively and consistently.

Identification of uncertain parameters, specification of ranges and probability distributions as well as dependencies between parameters are based on the experience which the experts gained from code validation or from experimental evidence. Therefore, expert judgement is used in this step of the uncertainty analysis. Compared with other uncertainty methods like AEA, CSAU and GE less judgement is necessary since no ranking of input parameters to reduce their number in order to cut computation cost is needed.
e) Combination of uncertainties

A *random sample* of size $n$ of sets of parameter values (simultaneous variations) is selected according to the specified subjective probability distributions and quantified dependencies and the corresponding code runs are performed. The GRS method relies on *actual code results without introducing fitted response surfaces or other approximations* like goodness of fit tests. One obtains the logically resulting distribution of selected key output parameters based on the specified state of knowledge of uncertain code parameters and models. The number of code calculation runs is given by the desired statistical tolerance limits, i.e. sufficient code calculations are performed in order to enable a statistically based evaluation of the uncertainty of the code results /1/, /6/, /7/. For example, the number of code runs required, to be 95% confident that at least 95% of the combined influence of all quantified uncertainties are below the tolerance limit, is 59 (or if 90% confidence level is sufficient then 45 code runs are necessary). The necessary number of code runs is not specified in any way for other methods like AEA, CSAU and GE.

In order to include uncertainties in scaling (beyond the demonstration in the validation process) additional uncertain model parameters can be introduced.

*Dependencies* between uncertain parameters are taken into account by deterministic relations, conditional distributions or association measures like correlation coefficients, etc.

*Compensating errors* of the computer code can be detected by analysing a sufficient number of key output parameters. Intermediate calculation results like heat transfer coefficients and final calculation results like cladding temperatures, for example, can be selected as key output parameters. The method can be applied to both continuous-valued (dependent on time) and single-valued (e.g. peak cladding temperature or vessel liquid level) output parameters. There is no chance to deal with compensating errors if the uncertainty of only one single-valued output parameter is considered (like in the CSAU and GE Methods). The number of required code calculations is independent of the number of selected key output parameters in the GRS Method. Therefore, a restriction to a small number of key output parameters is not necessary.
Cliff edge effects are adequately handled within the statistical framework only relying on a sufficient number of actual code calculation results to randomly selected parameter values. The method does not smooth out cliff edges by using fitted response surfaces like the CSAU and GE Methods.

Iteration is important for the GRS Method. Prior to an application for reactor plant calculations the method is used for an integral experiment which simulates the same or at least a similar accident scenario. If the experimental results are not bounded by the uncertainty statement the list of quantified uncertainties and their specified uncertainty ranges have to be revised. If the experimental values are bounded the uncertainty analysis for the plant will be the next step.

No expert judgement is used in the GRS Method for combination of uncertainties. Other methods need expert judgement for combination of uncertain parameter values to find the maximum and minimum values of the output parameter (which gives sometimes surprising results) and for finding the necessary number of code calculations (AEA Method). The CSAU and GE Methods use expert judgement to estimate a bias term of the output uncertainty to account for uncertainties not considered through the analysis, and to find the necessary number of code calculations.

f) Applications

The mathematical part of the GRS Method is almost completely developed. Some developmental work would be necessary to improve the basis for specification of the uncertainty ranges and distributions of the uncertain model parameters. Another activity would be to investigate the scaling effects in the code models.

So far the GRS Method has been applied to more than a dozen 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 No. 9, a blowdown with a PWR-type rod bundle consisting of 36 electrically heated rods of 3.66 m heated length. 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, and 1 parameter for noding change.

31 key output parameters were selected for determination of their uncertainty ranges and sensitivity measures, like time histories of temperatures, pressures, mass flow rates, void fractions, time of peak cladding temperatures (PCT), PCT, and total CPU time.

100 ATHLET code calculations were performed with randomly selected values of uncertain input parameters within their specified ranges and according to their probability distributions and quantified knowledge dependencies.

The uncertainty range of the calculated PCT was from 788°C to 1078°C (95%/95% statistical tolerance interval). 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. The temperature of 914°C is equal to the estimate of the 56% fractile of the resulting subjective probability distribution of the PCT (914°C will not be exceeded with a 56% subjective probability). The measured data for all the other output values were within their calculated uncertainty ranges as well.

Sensitivity measures like Standardized Rank Regression Coefficients were simultaneously derived from the sets of parameter values and the corresponding model output values to rank the uncertain parameters with respect to their contribution to output parameter uncertainty.

The following input parameters were found to have the largest contributions to the cladding temperature 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.

One of the most important lessons learned from 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 to be important after all. A very good knowledge of the validation and of the models of the specific computer code is essential.

g) Resources (staff time and computer time)

The uncertainty analysis for the SET required an effort of 1.5 person-years. The total CPU time was 20 hours on a CONVEX 3860 computer. The problem time for each run was 40 s. A total number of 100 ATHLET code calculations were performed (95%/95% statistical tolerance interval).

The resources planned to apply the method to a nuclear reactor plant is estimated to 7 person-years, including a prior uncertainty analysis of an integral experiment which simulates a similar accident scenario. A total number of 46 calculations is planned to be performed for the experiment and for the reactor plant as well (95%/90% statistical tolerance limit).

The main portion of resources are required for specification of individual input uncertainties and in performing the code runs. Most of the other steps are supported by a programme package. Considerable experience in quantifying the state of knowledge in uncertain parameters and models may reduce some of the resources needed for this quantification. Code speed-up and faster computers will reduce the computation time. A reduction of the necessary code runs is possible by reducing the required confidence in the quantitative uncertainty statement. The minimal number of code runs for the 95% fractile (denotes the value which will not be exceeded with 95% subjective probability) is:

59 at 95% confidence level,
45 at 90% confidence level,
32 at 80% confidence level,
28 at 75% confidence level,
24 at 70% confidence level,
14 at 50% confidence level.
h) Future developments and applications

Further development work is planned to improve the quantification of uncertain input parameters and to quantify the scaling effects. This will be performed in cooperation with the IPSN (Institut de Protection et de Sûreté Nucléaire), France.

The next application will be an uncertainty and sensitivity analysis of the integral experiment LSTF-SB-CL-18, which simulates a 5% break in the cold leg, loss of offsite power and ECC injection into the cold legs. After that an uncertainty and sensitivity analysis will be performed for a German reference PWR (1300 MWe) with 5% break in the cold leg, loss of offsite power and combined ECC injection (into hot and cold legs).

References

/1/ A. Forge, R. Pochard, H. Glaeser, W. Hobbhahn, E. Hofer, V. Teschendorff: Review study on uncertainty methods for thermal hydraulic computer codes; CEC Report, to be published

/2/ Thermohydraulics of emergency core cooling in light water reactors; CSNI Report No. 161, October 1989


/5/ Airy, Sir George Biddle: Theory of errors of observation; Macmillan, London, 1879


/7/ W.J. Conover: Practical non-parametric statistics; Wiley, New York, 1980

Uncertainty and Sensitivity Analysis of a Post-Experiment Calculation in Thermal Hydraulics

H. Glaeser, E. Hofer, M. Kloos, T. Skorek
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Forschungsgelände, D-85748 Garching, Germany

This uncertainty and sensitivity analysis accounts for modeling and parameter uncertainties originating from the experiment, the mathematical model, and the numerical solution algorithm.

Quantitative uncertainty statements are derived for code results that are either single-value quantities, like the peak cladding temperature, or continuous-value quantities, like the cladding temperature at a given position within the rod bundle. They do not resort to simplifications, like fitted response surfaces, etc., nor to approximations, like fitted distributions, but are obtained directly from the original code results, employing well-established concepts and tools from probability calculus and statistics. These statements account for the combined influence of modeling and parameter uncertainties as well as for the possible impact of the sampling error that is due to the limited number of code runs performed. Sensitivity measures to single- and continuous-value code results are presented and the benefit from consulting so-called correlation ratios, in addition to the more conventional regression-based measures, is explained and demonstrated.
1 THE ASSESSMENT QUESTION

A post-experiment calculation of the French OMEGA ROD BUNDLE Test No. 9\textsuperscript{1}, a blowdown with a PWR-type rod bundle, was analyzed, using the software system for uncertainty and sensitivity analysis SUSAN\textsuperscript{2}.\textsuperscript{3}

The test section consists of a rod bundle, simulating the reactor core, connected upstream and downstream through the 0.087 m diameter pipes to spherical plena. The bundle consists of 36 electrically heated rods of 3.66 m heated length, arranged in a 6 x 6 array. The plena simulate the primary circuit water volumes. Their inner diameter is 0.5 m. Convergent 0.198 m long break nozzles are connected horizontally to the spheres. The plena are connected upwards to the remaining parts of the loop from which they can be isolated by two quick-closing valves. The experiment starts with the establishment of a steady vertical upflow through the bundle. When the nominal con-

Fig. 1. The experimental facility\textsuperscript{1}
ditions are reached, the isolation valves are closed and the break rupture discs are destroyed, initiating the blowdown. The initial pressure is 13.4 MPa.

Several assessment questions were to be answered by the calculation. Among those of particular interest are:

1.) What is the time history of the cladding temperature in the high power region (i.e. the middle of the rod)?

2.) What is the overall maximum (or peak) cladding temperature?

![Diagram of rod channel with thermocouples](image)

Fig. 2. Schematic representation of the rod channel with positions of thermocouples. Level 7 is at 1.572 m above the bottom of the heated length and 0.258 m below mid-elevation of heated length.

2 THE ASSESSMENT MODEL

The experiment was calculated by the computer code ATHLET (Analyses of Thermal -Hydraulics of LEaks and Transients) developed by GRS. The range of applicability of this code comprises the whole spectrum of operational and abnormal transients,
small and intermediate leaks up to large breaks for pressurized water reactors, and boiling water reactors. For the present calculation, the five equation version was used which contains two mass conservation equations for steam and water, two energy equations, and one mixture momentum equation in combination with a drift-flux model to determine the relative velocity between steam and water.

Fig. 3. One of the two alternative nodalization schemes used in the ATHLET runs. The finer scheme differs in the noding of the two spherical plena.

3 THE ANALYSIS TOOL

The uncertainty and sensitivity analysis of the ATHLET application was performed with the software system SUSA\textsuperscript{23}.

SUSA supports the probabilistic modeling of parameter uncertainties. It offers a large selection of distribution types, takes truncations into account, derives distribution parameters from quantiles and plots the selected distributions. For the quantification of state of knowledge dependences, it accepts population measures of association, sample measure, conditional distributions, restrictions, complete dependence, functional relationships between uncertain parameters and provides scatter plots to illustrate the selected dependence quantification.
SUSA offers Simple Random Sampling and Latin Hypercube Sampling, derives distribution-free quantile estimates as well as statistical tolerance limits and performs statistical tests of distribution hypotheses and fitting of distributions to data. A choice of over 15 sensitivity measures is available for single- as well as continuous-value model output.

SUSA is panel-driven. A Personal Computer version (SUSA-PC) is in operation.

4 THE ELICITATION PROCESS

The procedure was an iterative one, extending over three meetings and involving intermediate interactions. All but few substantive experts (in-house) and one normative expert participated in the first meeting. The normative expert explained the assessment question, the aim of the analysis, sources of uncertainty, alternatives for the quantification of model uncertainty, the concept of subjective probability, the quantification and probabilistic modeling of the state of knowledge (thereby referring to the maximum entropy principle), bias from overconfidence and source of as well as means for quantification of state of knowledge dependence.

At the second meeting, eight substantive experts (model developers and code users) identified the phenomena relevant for the thermal hydraulic experiment and the corresponding models in the code. A list of potentially important uncertain model formulations and parameters was then compiled and agreed upon. According to their expertise, experts were assigned to individual phenomena, models and parameters (at most two experts each). They individually developed their state of knowledge quantifications after this meeting. Two substantive experts were determined to collect the information and to perform consistency checks. Subsequent discussions with and among the experts led to extensions of the compilation of uncertainties and to the narrowing or broadening of some parameter ranges. These changes were initiated by the consultation of additional literature and through some exploratory code runs showing results not in compliance with principles from physics.

Participants of the third meeting were the normative expert and the two substantive experts mentioned above. They discussed the compilation of uncertainties and corresponding state of knowledge quantifications item by item. Ranges, quantiles and distribution type were queried as well as the chances of having omitted any important state of knowledge dependences. Scatter plots were viewed to check how quantified
dependences would be represented in the sample to be drawn. A comparison with related analyses in the literature led to questions why some models or parameters were in the list or were omitted and why there were differences in state of knowledge quantifications. This comparison led to further extensions of the compilation and to minor revisions of state of knowledge quantifications.

A simple random sample of size 100 was then drawn from a joint subjective probability distribution satisfying the marginals and the dependence quantifications. The code was run for each of the 100 vectors of parameter values. It was found that the range specified for the correction factor to one of the uncertain model formulations (minimum film boiling temperature), and reduced already after the few exploratory runs, required further reduction as some of the results obtained for values from the lower end turned out to still violate principles from physics. Some of the 100 runs failed due to errors in the code which were subsequently corrected. Furthermore, consultation of the experimenters indicated that a potentially important uncertain parameter, describing the experimental procedure, had been overlooked. In fact, this parameter (switch-off time of the electrical heating) should later turn out to be the main contributor to uncertainty in the computed peak cladding temperature. The subsequently presented analysis results were obtained with a new simple random sample of size 100, drawn according to the revised list of uncertainties and state of knowledge quantifications, applying the corrected code.

5 POTENTIALLY IMPORTANT UNCERTAINTIES

A total of 60 potentially important uncertain parameters was identified by the procedure described above. They include 40 uncertain model parameters from those ATHLET models which describe relevant physical phenomena of the experiment, 11 parameters for selection of different correlations, 7 experiment specific parameters (3 properties of heater rod (Inconel), 2 parameters for closing times of hot and cold side isolation valves, 1 parameter for heater power, 1 parameter for heater power shutdown) and 2 parameters from the numerical solution algorithm (the local (in time) minimum absolute accuracy criterion for mass flow rates and two alternative spatial nodalizations). The uncertain model parameters were identified for the phenomena: Critical flow, evaporation, pressure loss (wall friction, form losses, momentum flux term), phase separation and mixture level (drift flux model, countercurrent flow
limitation), entrainment (drift flux model), and wall heat transfer. The final compilation of uncertainties was compared to those contained in the literature 5, 6, 7, 8.

Modeling uncertainties were expressed by additional uncertain parameters. They are already included in the total number of 60. These additional parameters are either:

a) uncertain correction factors to the preferred model formulation and / or
b) the index set of alternative model formulations.

Ideally, the alternatives would be mutually exclusive and the set complete. In practice, this will rarely be the case. Still, the state of knowledge of the appropriate model formulation may be expressed by a set of possibly applicable alternatives. In fact, for practical reasons, the actually implemented set will consist of a relatively small number of alternatives thought to be representative. For the purpose of sampling, their index set serves as the set of possibly applicable alternative values of an additional uncertain parameter.

For some model uncertainties (i.e. minimum film boiling) option a) was judged to be adequate, for others (i.e. momentum flux term) option b). In a number of instances (i.e. drift flux, stable film boiling, pool film boiling) options a) and b) were used, that means an uncertain correction factor was introduced for each individual alternative model formulation in the set.

6 STATE OF KNOWLEDGE QUANTIFICATIONS

Table 1 lists all of the uncertain parameters while Table 2 provides the density functions of the subjective probability distributions quantitatively expressing the states of knowledge of the parameters of Table 1.
Table 1. List of potentially important uncertain parameters

<table>
<thead>
<tr>
<th>NO</th>
<th>NAME</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>BLASI Model: critical discharge, turbulence factor</td>
</tr>
<tr>
<td>2</td>
<td>BLASI Model: critical discharge, Weisbach-Darcy friction factor, FD</td>
</tr>
<tr>
<td>3</td>
<td>Evaporation, number of vapour bubbles per unit volume</td>
</tr>
<tr>
<td>4</td>
<td>Evaporation, number of liquid droplets per unit volume</td>
</tr>
<tr>
<td>5</td>
<td>Evaporation, multiplication factor for eddy conductivity</td>
</tr>
<tr>
<td>6</td>
<td>Friction factor, pipe</td>
</tr>
<tr>
<td>7</td>
<td>Friction factor, rod bundle</td>
</tr>
<tr>
<td>8</td>
<td>Pressure drop, momentum term at rod bundle, hot leg</td>
</tr>
<tr>
<td>9</td>
<td>Pressure drop, form loss, plenum inlet</td>
</tr>
<tr>
<td>10</td>
<td>Pressure drop, form loss, plenum outlet</td>
</tr>
<tr>
<td>11</td>
<td>Pressure drop, form loss, Venturi nozzle</td>
</tr>
<tr>
<td>12</td>
<td>Pressure drop, form loss, flow meter (forward, backward)</td>
</tr>
<tr>
<td>13</td>
<td>Pressure drop, form loss, bends</td>
</tr>
<tr>
<td>14</td>
<td>Pressure drop, form loss, core inlet (forward, backward)</td>
</tr>
<tr>
<td>15</td>
<td>Pressure drop, form loss, core spacer (forward, backward)</td>
</tr>
<tr>
<td>16</td>
<td>Pressure drop, form loss, core mixing grid (forward, backward)</td>
</tr>
<tr>
<td>17</td>
<td>Pressure drop, form loss, core outlet (forward, backward)</td>
</tr>
<tr>
<td>18</td>
<td>Pressure drop, form loss, diffusor with side branch: backward</td>
</tr>
<tr>
<td>19</td>
<td>Pressure drop, form loss, break nozzle</td>
</tr>
<tr>
<td>20</td>
<td>Pressure drop, form loss, valve hot leg</td>
</tr>
<tr>
<td>21</td>
<td>Pressure drop, two-phase flow multiplier</td>
</tr>
<tr>
<td>22</td>
<td>Pressure drop, correction factor for two-phase flow multiplier</td>
</tr>
<tr>
<td>23</td>
<td>Pressure drop, relative roughness e/D, pipes &amp; plena</td>
</tr>
<tr>
<td>24</td>
<td>Pressure drop, relative roughness e/D, rod bundle</td>
</tr>
<tr>
<td>25</td>
<td>Drift-Flux-Model, pipe vertical</td>
</tr>
<tr>
<td>26</td>
<td>Correction factor for drift flux correlation, pipe vertical</td>
</tr>
<tr>
<td>27</td>
<td>Drift-Flux-Model, pipe horizontal</td>
</tr>
<tr>
<td>28</td>
<td>Correction factor for Flooding Based Drift-Flux Model in horizontal pipe</td>
</tr>
<tr>
<td>29</td>
<td>Drift-Flux-Model, plenum</td>
</tr>
<tr>
<td>30</td>
<td>Correction factor for drift flux correlations in plena</td>
</tr>
<tr>
<td>31</td>
<td>Drift-Flux-Model, bundle</td>
</tr>
<tr>
<td>32</td>
<td>Correction factor for drift flux correlations in bundle</td>
</tr>
<tr>
<td>33</td>
<td>Drift-Flux-Model, side branch at plenum</td>
</tr>
<tr>
<td>34</td>
<td>Correction factor for side branch at plenum</td>
</tr>
<tr>
<td>35</td>
<td>M-constant in Wallis correlation</td>
</tr>
<tr>
<td>36</td>
<td>C-constant in Wallis correlation</td>
</tr>
<tr>
<td>37</td>
<td>M-constant in Kutateladze correlation</td>
</tr>
<tr>
<td>38</td>
<td>C-constant in Kutateladze correlation</td>
</tr>
<tr>
<td>39</td>
<td>Correction factor for forced convection on water</td>
</tr>
<tr>
<td>40</td>
<td>Correction factor for natural convection on water</td>
</tr>
<tr>
<td>41</td>
<td>Heat transfer, model for forced convection on vapour</td>
</tr>
<tr>
<td>42</td>
<td>Heat transfer, correction factor for forced convection on vapour</td>
</tr>
<tr>
<td>43</td>
<td>Heat transfer, correction factor for nucleate boiling</td>
</tr>
<tr>
<td>44</td>
<td>Heat transfer, model for critical heat flux</td>
</tr>
<tr>
<td>45</td>
<td>Heat transfer, correction factor for critical heat flux</td>
</tr>
<tr>
<td>46</td>
<td>Heat transfer, correction factor for Groeneveld correlation for min. film boiling and rewetting temperature</td>
</tr>
<tr>
<td>47</td>
<td>Heat transfer, rewetting temperature correlation</td>
</tr>
<tr>
<td>48</td>
<td>Heat transfer, stable film boiling correlation</td>
</tr>
<tr>
<td>49</td>
<td>Heat transfer, correction factor for stable film boiling</td>
</tr>
<tr>
<td>50</td>
<td>Heat transfer, model for pool film boiling</td>
</tr>
<tr>
<td>51</td>
<td>Heat transfer, correction factor for pool film boiling</td>
</tr>
<tr>
<td>52</td>
<td>Correction factor for thermal conductivity</td>
</tr>
<tr>
<td>53</td>
<td>Correction factor for specific heat capacity</td>
</tr>
<tr>
<td>54</td>
<td>Correction factor for density</td>
</tr>
<tr>
<td>55</td>
<td>Time of valve closing, cold leg</td>
</tr>
<tr>
<td>56</td>
<td>Time of valve closing, hot leg</td>
</tr>
<tr>
<td>57</td>
<td>Core power</td>
</tr>
<tr>
<td>58</td>
<td>Local minimum absolute accuracy criterion for mass-flow rates</td>
</tr>
<tr>
<td>59</td>
<td>Spatial nodalization</td>
</tr>
<tr>
<td>60</td>
<td>Time delay of heating switch off</td>
</tr>
</tbody>
</table>
Table 2. State of knowledge quantifications

<table>
<thead>
<tr>
<th>DISTRIBUTION</th>
<th>CODE</th>
<th>DISTRIBUTION PARAMETERS</th>
<th>NUMBER OF POINTS</th>
<th>PROBABILITIES</th>
</tr>
</thead>
<tbody>
<tr>
<td>Discrete Sir.</td>
<td>1</td>
<td>4</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td></td>
<td>4</td>
<td></td>
<td>0.6</td>
<td>0.1</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.3</td>
<td></td>
</tr>
<tr>
<td>Uniform Sir.</td>
<td>5</td>
<td>3</td>
<td>1.0</td>
<td>2.0</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
<tr>
<td>Discrete Sir.</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>3</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td></td>
<td>0.6</td>
<td>0.1</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
<tr>
<td>Uniform Sir.</td>
<td>5</td>
<td>2</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
<tr>
<td>Discrete Sir.</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td></td>
<td>0.25</td>
<td>0.5</td>
</tr>
<tr>
<td>Uniform Sir.</td>
<td>5</td>
<td>1</td>
<td>0.6</td>
<td>1.3</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
<tr>
<td>Discrete Sir.</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td></td>
<td>0.6</td>
<td>0.1</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
<tr>
<td>Uniform Sir.</td>
<td>5</td>
<td>1</td>
<td>0.6</td>
<td>1.3</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
<tr>
<td>Discrete Sir.</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>4</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td></td>
<td>0.6</td>
<td>0.1</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
<tr>
<td>Uniform Sir.</td>
<td>5</td>
<td>1</td>
<td>0.6</td>
<td>1.3</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
<tr>
<td>Discrete Sir.</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>4</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td></td>
<td>0.6</td>
<td>0.1</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
<tr>
<td>Uniform Sir.</td>
<td>5</td>
<td>1</td>
<td>0.6</td>
<td>1.3</td>
</tr>
<tr>
<td></td>
<td>0.1</td>
<td></td>
<td>0.1</td>
<td></td>
</tr>
</tbody>
</table>

Note: The table continues with similar entries for different types of distributions and their respective parameters.
Table 2. State of knowledge quantifications (continued)

<table>
<thead>
<tr>
<th>Par. No.</th>
<th>Index Value</th>
<th>Alternative Models</th>
</tr>
</thead>
<tbody>
<tr>
<td>8-1</td>
<td>-1</td>
<td>Momentum term from hot leg only</td>
</tr>
<tr>
<td></td>
<td>0</td>
<td>Momentum term not computed</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Momentum term in both directions</td>
</tr>
<tr>
<td>21</td>
<td>1</td>
<td>Martineili-Nelson Model with constant friction factors</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Martineili-Nelson Model with computed friction factors</td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>Chisholm Model with computed friction factors</td>
</tr>
<tr>
<td>25</td>
<td>1</td>
<td>Flooding Based Drift Model</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Explicit Drift Model with Vicedenz-Correlation</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>Explicit Drift Model with Wilson-Correlation</td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>Explicit Drift Model with Ishii-Correlation</td>
</tr>
<tr>
<td>29</td>
<td></td>
<td>see parameter 25</td>
</tr>
<tr>
<td>31</td>
<td>1</td>
<td>Flooding Based Drift Model</td>
</tr>
<tr>
<td></td>
<td>3, 4</td>
<td>see parameter 25</td>
</tr>
<tr>
<td>41</td>
<td>1</td>
<td>Ditius-Boelter-Correlation</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Mc Elgot-Correlation</td>
</tr>
<tr>
<td>44</td>
<td>0</td>
<td>Minimum value</td>
</tr>
<tr>
<td></td>
<td>1</td>
<td>Westinghouse W-3</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Hench-Levy</td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>Blasi</td>
</tr>
<tr>
<td>47</td>
<td>2</td>
<td>Groeneveld-Correlation</td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>Groeneveld-Correlation with quality criterion</td>
</tr>
<tr>
<td></td>
<td>5</td>
<td>Groeneveld-Correlation with correction for small enthalpy-qualities</td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>Groeneveld-Correlation with quality criterion and correction for small enthalpy-qualities</td>
</tr>
<tr>
<td>48</td>
<td>1</td>
<td>Modified Dougall-Rhonenow-Correlation</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Condle-Bengston IV-Correlation</td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>Groeneveld 5.9-Correlation</td>
</tr>
<tr>
<td>50</td>
<td>1</td>
<td>Berenson-Correlation</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Bromley-Correlation</td>
</tr>
<tr>
<td>59</td>
<td>1</td>
<td>Coarse noding in spheres</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>Fine noding in spheres at the break line connection</td>
</tr>
</tbody>
</table>

7 STATE OF KNOWLEDGE DEPENDENCES

Several instances of state of knowledge dependence were identified by the experts. They are listed in Table 3 together with the chosen option for quantification.

From the options offered by SUSA, population measures of association (i.e. parameters 3 and 5), complete dependence (i.e. parameters 29 and 25), and conditional distributions (i.e. for parameter 22 under the condition of the value selected for parameter 21) were used. Tables 1 to 3 contain output from the documentation support offered by SUSA.
### Table 3. State of knowledge dependence quantifications

<table>
<thead>
<tr>
<th>Code</th>
<th>Name</th>
<th>NO. of PL.</th>
<th>Parameters</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>BETA</td>
<td>2*1</td>
<td>0,0.5,0.8,0.5</td>
</tr>
<tr>
<td>2</td>
<td>GAMMA</td>
<td>2*1</td>
<td>0,0.5,0.8,0.5</td>
</tr>
<tr>
<td>3</td>
<td>EXTR.1</td>
<td>2*1</td>
<td>0,0.5,0.8,0.5</td>
</tr>
<tr>
<td>4</td>
<td>EXTR.2</td>
<td>2*1</td>
<td>0,0.5,0.8,0.5</td>
</tr>
</tbody>
</table>

#### Conditional Distribution

<table>
<thead>
<tr>
<th>Code</th>
<th>Name</th>
<th>NO. of PL.</th>
<th>Parameters</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>BETA</td>
<td>2*1</td>
<td>0,0.5,0.8,0.5</td>
</tr>
<tr>
<td>2</td>
<td>GAMMA</td>
<td>2*1</td>
<td>0,0.5,0.8,0.5</td>
</tr>
<tr>
<td>3</td>
<td>EXTR.1</td>
<td>2*1</td>
<td>0,0.5,0.8,0.5</td>
</tr>
<tr>
<td>4</td>
<td>EXTR.2</td>
<td>2*1</td>
<td>0,0.5,0.8,0.5</td>
</tr>
</tbody>
</table>

#### Pair Dependence

<table>
<thead>
<tr>
<th>Code</th>
<th>NO.</th>
<th>CODE + LIST OF DISTRIBUTIONAL PARAMETERS</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>3</td>
<td>1.0, 0.75, 1.25, 1.75</td>
</tr>
<tr>
<td>2</td>
<td>3</td>
<td>1.0, 0.75, 1.25, 1.75</td>
</tr>
<tr>
<td>3</td>
<td>3</td>
<td>1.0, 0.75, 1.25, 1.75</td>
</tr>
</tbody>
</table>

### Table 4. Pale dependence quantifications

<table>
<thead>
<tr>
<th>Code</th>
<th>NO.</th>
<th>CODE + LIST OF DISTRIBUTIONAL PARAMETERS</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>3</td>
<td>1.0, 0.75, 1.25, 1.75</td>
</tr>
<tr>
<td>2</td>
<td>3</td>
<td>1.0, 0.75, 1.25, 1.75</td>
</tr>
<tr>
<td>3</td>
<td>3</td>
<td>1.0, 0.75, 1.25, 1.75</td>
</tr>
</tbody>
</table>
Expressing the state of knowledge by subjective probability at the parameter and sub-model level entails a subjective probability distribution for each of the code results. These latter distributions quantitatively express the logically resulting state of knowledge at the assessment level. For complex codes like ATHLET, they remain unknown, however. 100 60-tuples of parameter values were selected at random from a joint subjective probability distribution satisfying the experts' state of knowledge quantifications. ATHLET was run for each of those to obtain the corresponding alternative assessments. They constitute a random sample from the respective unknown distributions, and quantitative uncertainty statements can be immediately derived from this sample. Figure 4 shows the 100 alternative time histories of the void fraction at the cold leg while Figure 5 presents those of the corresponding cladding temperatures at level 7.

Fig. 4. Alternative time histories of the void fraction at the cold leg, obtained with the 100 randomly selected vectors of parameter values.
Fig. 5. Alternative time histories of the cladding temperature at level 7, obtained with the 100 randomly selected vectors of parameter values

9 UNCERTAINTY STATEMENTS FOR SELECTED CODE RESULTS

Figure 6 shows the continuous connections of the upper, respectively lower, end points of local (95, 95) statistical tolerance intervals (solid lines). They say that one can be at least 95 % confident that at least 95 % of the combined influence of all quantified parameter and modeling uncertainties are between the pair of temperature values to be read from the two curves at each point in time. This statement does also account for the possible impact of the sampling error which is due to the fact that only a sample of size 100 was selected from the unknown subjective probability distribution of the cladding temperature at the respective point in time. The time history obtained with the best estimate values of the uncertain parameters (default and recommended values in the code and best estimate values for initial and boundary conditions in the experiment as well as for material properties) is also shown in Figure 6. Figure 7 presents the continuous connections of the local minimum and maximum among the computed and the measured temperature values at level 7.
Fig. 6. At any point in time at least 95% of the combined influence of all quantified modeling and parameter uncertainties are between the lower and upper curve, at a confidence level of at least 95%. The time history obtained with the best estimate model formulations and parameter values lies between those limits.

Fig. 7. Computed and measured temperature values (minimum and maximum values at level 7). The thermal inertia of the thermocouples is one reason why the measurements are not within the band of the computed cladding temperature.
Figure 8 presents the empirical distribution of the 100 alternative peak cladding temperatures (PCT). The highest measured PCT is 780°C. However, due to the thermal inertia of the thermocouples (thermocouple diameter 1mm compared with cladding thickness in the hottest region 0.5 mm), the measured PCTs have to be corrected. The corrected measured PCT is 914°C which is equal to the estimate of the 56% quantile of the resulting subjective probability distribution of the PCT.

![Graph showing empirical distribution of peak cladding temperatures.]

Fig. 8. Empirical distribution of the peak cladding temperature values obtained with the 100 randomly selected vectors of parameter values. At least 95% of the combined influence of all quantified modeling and parameter uncertainties are within the interval indicated on the abscissa, at a confidence level of at least 95%.

In total, a number of 31 key results were selected to determine uncertainty ranges and sensitivity measures. These are single-value results (1 peak cladding temperature, 1 time when peak clad temperature occurs, 1 total CPU time) and continuous-value results (7 clad temperatures, 6 fluid temperatures, 6 pressures, 6 mass flow rates, 3 void fractions).
10 SENSITIVITY STATEMENTS FOR SELECTED CODE RESULTS

Figure 9 shows standardized rank regression coefficients as sensitivity measures over the time span of largest uncertainty in the computed cladding temperature. Additionally the $R^2$ value is presented, indicating that in general well over 70% of the variability in the rank-transformed temperature values is explained by these measures. According to Figure 9, the main contributions to uncertainty are due to parameters 1 (critical flow model), 3, 5 (evaporation model), 26, 30, 31 (drift flux model), 43, 49 (heat transfer model), 57, 60 (the experiment).

A positive sign of the measure means that parameter value and cladding temperature tend to move in the same direction while in case of a negative sign they tend to move in opposite directions. Partial rank correlation coefficients were also computed and provided additional insight particularly where state of knowledge dependence was involved.

---

Legends:
- PRR. 1 15 31 46
- PRR. 2 16 32 47
- PRR. 3 17 34 48
- PRR. 4 18 35 49
- PRR. 5 19 36 50
- PRR. 6 20 37 51
- PRR. 7 21 38 52
- PRR. 8 22 39 53
- PRR. 9 23 40 54
- PRR. 10 24 41 55
- PRR. 11 25 42 56
- PRR. 12 26 43 57
- PRR. 13 27 44 58
- PRR. 14 28 45 59
- PRR. 15 29 46 60

Fig. 9. Standardized rank regression coefficients (Spearman) as sensitivity measures with respect to the cladding temperature at level 7 over the relevant time interval.
Parameters 8, 21, 25, 29, 31, 44, 47, 48 represent modeling uncertainties in the form of index values of at least three alternative model formulations. Different regression coefficients may be obtained, depending on the order in which the alternative models are indexed. A measure that is less susceptible to this index ordering is the so-called correlation ratio

\[ r(Y, X) = \frac{\text{Var} \{E(Y/X)\}}{\text{Var} \{Y\}} \]  

derived from the well-known relationship

\[ \text{Var} \{Y\} = E\{\text{Var} \{Y/X\}\} + \text{Var} \{E\{Y/X\}\} \]

where \( \text{Var} \) and \( E \) stand for variance and expected value, respectively. In case of independence between \( Y \) and \( X \), the first term on the right-hand side of equation (2) equals \( \text{Var} \{Y\} \) and the second term vanishes and vice versa in case of complete dependence. Obviously, the measure is always positive and therefore not capable of indicating directions of the parameter influence.

Figure 10 presents correlation ratios as sensitivity measures, obtained via a discretization of the parameter and temperature axis into ten classes. According to Figure 10, the main contributions to uncertainty are due to parameters 1, 46, 48 and 60.

![Diagram](image)

**Fig. 10.** Correlation ratios (see (1) above) as sensitivity measures with respect to the cladding temperature at level 7 over the relevant time interval.
Like ordinary correlation coefficients, correlation ratios may not give an adequate ranking of the contributions to uncertainty if state of knowledge dependence is involved. Almost all of the parameters, additionally indicated as important by the measure used in Figure 9, are subject to this kind of dependence. On the other hand, the cladding temperature at level 7 exhibits an association (see Figure 11) with parameter 46 which is not adequately expressed by regression coefficients but well captured by correlation ratios. The same applies to parameter 48 (see Figure 12).

Figure 13 shows the correlation ratios for the peak cladding temperature (PCT). According to this figure, parameters 46 and 60 contribute most to the uncertainty expressed by the spread of the empirical distribution function of Figure 8.

Fig. 11. Scatter plot of the cladding temperature at level 7 and time 4 sec., with respect to uncertain parameter 46 (correction factor to Groeneveld-Stewart-Correlation for the minimum film boiling temperature). Parameter 46 determines when to switch to the stable film boiling correlation.
Fig. 12. Scatter plot of the cladding temperature at level 7 and time 10.5 sec., with respect to uncertain parameter 48 (alternative stable film boiling correlations)

Fig. 13. Correlation ratios as sensitivity measures with respect to peak cladding temperature
A scatter plot indicates that the largest computed PCTs are obtained for some of the lower values of parameter 46. This association is, however, not sufficient to lead to an outstanding standardized rank regression coefficient, but the correlation ratios do account for it. In fact, the cluster of cladding temperature time histories beginning to rise at about 3 seconds to some of the highest cladding temperatures in Figure 5 are due to these lower values of parameter 46. In these computations, film boiling is brought about already by the pressure drop immediately following the destruction of the rupture disks. The reason is the pressure dependence in the Groeneveld-Stewart minimum film boiling temperature correlation. The minimum film boiling temperature drops below the calculated cladding temperatures of the heater rod due to the decrease in system pressure. The standardized rank regression coefficients top rank parameters 3, 5 and 60 and achieve an $R^2$-value of about 85%. Parameters 3 and 5, are strongly negative state of knowledge dependent.

Fig. 14. Standardized rank regression coefficients (Spearman) as sensitivity measures with respect to peak cladding temperature.
For code developers, it is of some interest to receive hints at any algorithmic and modeling inefficiencies. Therefore, sensitivity measures were derived for the central processor unit (CPU) time required for each run. They are presented in Figure 16 while Figure 15 shows the empirical distribution function of the CPU-time used on a CONVEX C3840 machine. A little more than 10% of the runs required more than 1000 seconds, the maximum lying over 3000 seconds while 650 seconds is close to the median of the distribution. The correlation ratios identify the modeling uncertainties (parameters 25 and 31), expressed by alternative model formulations, as the main contributors to the spread of the empirical distribution of the CPU time. These parameters represent uncertainties in the drift flux modeling. Clearly, the respective drift flux models seem to have the potential of influencing the required CPU time considerably. This did not surprise the code developers although they expected the contributions from uncertain parameters of the numerical solution algorithm, like the local absolute accuracy criterion and alternative spatial nodalization schemes (parameters 58 and 59) to receive the highest ranking with respect to CPU time.
Fig. 15. Cumulative distribution of the individual CPU-time requirements of the 100 model runs, performed on a CONVEX C3840 machine.

Fig. 16. Correlation ratios as sensitivity measures, with respect to the CPU-time required per run.
11 CONCLUSIONS

To our knowledge, this is for the first time that a loss of coolant assessment is analyzed by a fully probabilistic approach considering various modeling uncertainties and not resorting to any simplifying approximations like in\textsuperscript{6,7,8}. Correspondingly, the derived uncertainty statements quantify the combined influence of modeling, parameter, and experimental uncertainties and account for the possible influence of the sampling error that is due to the fact that the statements are obtained from a random sample of size 100 only. The latter cannot be quantified for a Latin Hypercube sample.

The sensitivity measures indicate where the state of knowledge should be improved if the uncertainty of the respective code result is to be reduced most effectively. Altogether a total of 31 single- and continuous-value code results was analyzed.

With respect to the cladding temperature at level 7, it turns out that specifics of the experiment need to be known more precisely, followed by phenomena from the area of heat transfer.

Correlation ratios proved to be a valuable addition to the set of sensitivity measures, particularly in the presence of strongly non-monotonic relationships which may, for instance, come about by the order in which alternative model formulations, expressing modeling uncertainty, are indexed. Generally one should, however, not rely on only this type of sensitivity measure. Standardized regression coefficients can reveal additional useful sensitivity information in case of state of knowledge dependencies between uncertain parameters.

Last, but not least, the analysis contributed to the quality assurance process of the code. Running the code for many sets of parameter values and model formulations, selected from the combined multidimensional uncertainty range, may certainly be considered as a thorough "robustness" check. So far, almost all of our uncertainty and sensitivity analyses, performed for applications of various codes, have initiated improvements to the codes. It should also be mentioned that the present analysis added to a better understanding of the effect of values from the lower end of the uncertainty range of the minimum film boiling temperature on the time history of the cladding temperature.
Encouraged by the experience gained, an uncertainty and sensitivity analysis of a code application to an integral experiment (5% leak) will be our next step to the analysis of full size power plant assessments.

References

/1/ C. Chauliac, OMEGA Depressurization d'un ensemble comportant une grappe de 36 barreaux chauffants, Note TT/SETRE/82-26, Tome 1, Décembre 1982.


CEA/IPSN

UNCERTAINTY METHOD

Set up by GTTh members:

M. Reocreux, M. Champ, R. Pochard, P. Probst, C. Renault

Presentation by:

R. Pochard
A. Ounsi
A. De Crecy
WHY WE NEED UNCERTAINTY EVALUATION

- TO USE BEST ESTIMATE CODES
  - in licensing process
  - for procedure evaluation
  - beyond DBA studies

- TO ANSWER TO THE QUESTION
  How good is good enough?

  if the code is not good enough
  - needs for code development
  - needs for code improvements

  if the code is good enough
  - only code maintenance programme
**POSITION OF THE PROBLEM**

**CODE**

- Parameters: $X_1, \ldots, X_p$
- Code execution: $A(V, X_1, \ldots, X_p) = 0$
- State vector: $V = V(X_1, \ldots, X_p)$
- Response: $Y = R(V)$
- $Y = F(X_1, \ldots, X_p)$

**UNCERTAINTIES ON PARAMETERS.**

- $f_1$: PDF of $X_1$ on $[a_1, b_1]$
- $f_p$: PDF of $X_p$ on $[a_p, b_p]$

**UNCERTAINTY ON RESPONSE $Y$.**

- Determination of:
  - Range: $[a_y, b_y]$
  - PDF: $f_Y$

**SENSITIVE PARAMETERS**

**PROBLEM**

- Given: $\Delta X_1, \ldots, \Delta X_p$
- How to get: $\Delta Y = \Delta Y(\Delta X_1, \ldots, \Delta X_p)$
• DETERMINATION OF PARAMETER UNCERTAINTIES.

• STATISTICS FRAMEWORK.

• SAMPLING METHODS.

• CALCULATION OF CONFIDENCE INTERVAL.

• GOODNESS OF FIT TESTS.

• SENSITIVITY ANALYSIS.
  • STATISTICAL ANALYSIS
  • ASM METHOD

• 1st APPROACH.
  • CALCULATION OF A RESPONSE SURFACE
    • of first order: standard linear regression.
    • of second order: use of ASM.
  • MONTE-CARLO SIMULATION OF RESPONSE PDF

• 2nd APPROACH.
  • RESPONSE SAMPLING (SIMPLE OR LHS).
  • CALCULATION OF A CONFIDENCE INTERVAL.
  • GOODNESS OF FIT TESTING.
  • PDF ESTIMATIONS.
• RESPONSE SAMPLE OF SIZE N : Y1, ..., YN.

\[ m = \min\{Y_j, j=1, ..., N\} \quad M = \max\{Y_j, j=1, ..., N\} \]

\( \bar{Y} \) et \( S_y \) : mean and standard deviation of \( Y \).

\( 0 < \gamma < 1, 0 < S < 1 \).

• WILKS FORMULA.

If \( N \geq N(\gamma, S) \) Then

\([m, M] = \) two sided \( \gamma \)-fractile at confidence level \( S \).

• EXAMPLES OF VALUES OF \( N(\gamma, S) \)

<table>
<thead>
<tr>
<th>S</th>
<th>( \gamma )</th>
<th>0.50</th>
<th>0.90</th>
<th>0.95</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.5</td>
<td>3</td>
<td>17</td>
<td>34</td>
<td></td>
</tr>
<tr>
<td>0.8</td>
<td>5</td>
<td>29</td>
<td>59</td>
<td></td>
</tr>
<tr>
<td>0.90</td>
<td>7</td>
<td>38</td>
<td>77</td>
<td></td>
</tr>
<tr>
<td>0.95</td>
<td>8</td>
<td>46</td>
<td>93</td>
<td></td>
</tr>
</tbody>
</table>

• TESTING THE FIT TO A KNOWN PROBA LAW

\( \chi^2 \) test:
comparison of PDFs

Kolmogorov-Smirnov test:
comparison of CPDFs

• TOLERANCE LIMIT FACTOR.

If there is fit to \( \ell \) law:

\[ k = k(N, \gamma, S, \ell) \]

such that

\[ [\bar{Y} - kS_y, \bar{Y} + kS_y] = \) two sided \( \gamma \)-fractile at confidence level \( S \). \]
GOODNESS OF FIT TESTS

khi2 test

SENSITIVITY ANALYSIS
(with regression)

• STANDARD REGRESSION COEFFICIENTS:

\[
\frac{Y - \bar{Y}}{S_Y} = \sum \alpha_i \left( \frac{X_i - \bar{X}}{S_i} \right)
\]

\[
\alpha_i^*
\]

d = fraction of response standard deviation explained by X_i.

• ELIMINATION OF COMBINED EFFECTS.

Partial Linear Correlation Coefficients

linear dependency

****

Partial Rank Correlation Coefficients
(Spearman)

monotous dependency

Kolmogorov-Smirnov test
ONE N-SAMPLE PER PARTIAL DERIVATIVE:

\[ \frac{\partial Y}{\partial X_1}, \ldots, \frac{\partial Y}{\partial X_n} \]

\[ \frac{\partial Y}{\partial X_p}, \ldots, \frac{\partial Y}{\partial X_p} \]

PARTIAL DERIVATIVES RANGE:

\[ \left( \frac{\partial Y}{\partial X_1} \right)_{\text{min}}, \left( \frac{\partial Y}{\partial X_1} \right)_{\text{max}} \]

SENSITIVITY MEASURES:

\[ \left( \frac{\partial Y}{\partial X_1} \right)_{\text{moyen}} \times (b_i - a_i) \]

APPLICATION OF ASM

- Deterministic quantification of sensitivities.

- Cross-analysis with statistics.

- Optimized calculation of 2nd order response surface.

- No limit to the number of parameters.
**TEST CASE DESCRIPTION**

- **CODE USED**
  - SIMPLIFIED VERSION OF CATHARE 2 V1.4
  - 6 EQUATIONS MODEL (PIPE MODULE)
  - FULLY IMPLICIT SCHEME FOR THERMOHYDRAULICS
  - IMPlicit COUPLING OF THERMOHYDRAULICS WITH HEAT CONDUCTION
  - SET OF SIMPLIFIED PHYSICAL LAWS

- **PHYSICAL TEST**
  - DEPRESSURIZATION OF A VERTICAL PIPE (VERTICAL CANON TEST 22)

**TEST PARAMETERS**

- Length: 4.582 m
- Pipe diameter: 10 cm
- Break diameter: 5 mm
- Initial pressure: 57 bar
- Initial subcooling: 40 °C
- Initial water mass: 30 kg

**Comparison ASM / Brute Force**

- **Response**
  - remaining mass at the end of the test (t=400s)
  - reference value: M = 13.987 kg

- **Local uncertainty evaluation**
  - increase of 10% for each parameter $X_i$
  - Brute Force method: $\Delta R = R(X_i + \Delta X_i) - R(X_i)$
  - ASM method: $\Delta R = \Delta X_i \times \frac{\partial R}{\partial X_i}$

- **Results**
  - Response variation larger than 100g for 4 parameters:
    - $T_{oi}$
    - $T_{ob}$
  - Break diameter,
  - Wall heat capacity.

  Same sensitive parameters for ASM and Brute Force
  Similar variation with ASM and Brute Force
Assumed parameter uncertainty

<table>
<thead>
<tr>
<th>Parameters</th>
<th>ID</th>
<th>Uncertainties (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Break diameter</td>
<td>D</td>
<td>±1.</td>
</tr>
<tr>
<td>Wall capacity</td>
<td>C_{ww}</td>
<td>±1.25</td>
</tr>
<tr>
<td>Wall conductivity</td>
<td>λ_{w}</td>
<td>±3.</td>
</tr>
<tr>
<td>Liquid heat transfer coefficient</td>
<td>H_{ld}</td>
<td>±50.0</td>
</tr>
<tr>
<td>Nucl. boiling heat transfer coeff</td>
<td>H_{nb}</td>
<td>±50.0</td>
</tr>
<tr>
<td>Gas heat transfer coefficient</td>
<td>H_{ga}</td>
<td>±50.0</td>
</tr>
<tr>
<td>Transition temperature</td>
<td>T_{tr}</td>
<td>±50.0</td>
</tr>
<tr>
<td>Liquid to interface heat transfer</td>
<td>Q_{LE}</td>
<td>±50.0</td>
</tr>
<tr>
<td>Gas to interface heat transfer</td>
<td>Q_{VE}</td>
<td>±50.0</td>
</tr>
<tr>
<td>Liquid/Wall friction</td>
<td>C_{l}</td>
<td>±50.0</td>
</tr>
<tr>
<td>Gas/Wall friction</td>
<td>C_{g}</td>
<td>±50.0</td>
</tr>
<tr>
<td>Interfacial friction Tobb</td>
<td>T_{obb}</td>
<td>±50.0</td>
</tr>
<tr>
<td>Interfacial friction Toa</td>
<td>T_{oa}</td>
<td>±50.0</td>
</tr>
<tr>
<td>Entrainment</td>
<td>E</td>
<td>±50.0</td>
</tr>
</tbody>
</table>

Assumed PDF: uniform law

- **Statistical level**
  \[
  \begin{align*}
  \gamma &= 0.95 \\
  S &= 0.80
  \end{align*}
\]
  \(\Rightarrow\) sample size \(N = 59\) (two-sided Wilks formula)

- **First approach**

<table>
<thead>
<tr>
<th></th>
<th>Confidence interval</th>
</tr>
</thead>
<tbody>
<tr>
<td>Experiment</td>
<td>[13.300, 14.700]</td>
</tr>
<tr>
<td>Wilks formula</td>
<td>[13.224, 15.095]</td>
</tr>
<tr>
<td>Goodness of fit test</td>
<td>[12.902, 15.225]</td>
</tr>
</tbody>
</table>

- **Second approach**

  **Monte-Carlo simulation**

  - Surface response of the 1st order
determination coefficient = 0.98,
  \[
  Y \leq 14,912 \quad \text{at} \quad 95\%
  \]

  - Surface response of the 2nd order
determination coefficient = 0.98,
  \[
  Y \leq 15,110 \quad \text{at} \quad 95\%
  \]
● Parameter ranking

<table>
<thead>
<tr>
<th>Parameters</th>
<th>D</th>
<th>C_{mw}</th>
<th>\lambda_w</th>
<th>H_{cl}</th>
<th>H_{nb}</th>
<th>H_{co}</th>
<th>T_{tr}</th>
<th>Q_{le}</th>
<th>Q_{ve}</th>
<th>C_{l}</th>
<th>C_{g}</th>
<th>T_{obb}</th>
<th>T_{oa}</th>
<th>E</th>
</tr>
</thead>
<tbody>
<tr>
<td>Standard regression</td>
<td>2</td>
<td>14</td>
<td>10</td>
<td>9</td>
<td>3</td>
<td>12</td>
<td>7</td>
<td>5</td>
<td>13</td>
<td>6</td>
<td>8</td>
<td>1</td>
<td>11</td>
<td>4</td>
</tr>
<tr>
<td>Partial regression</td>
<td>2</td>
<td>14</td>
<td>10</td>
<td>9</td>
<td>3</td>
<td>12</td>
<td>7</td>
<td>5</td>
<td>13</td>
<td>6</td>
<td>8</td>
<td>1</td>
<td>11</td>
<td>4</td>
</tr>
<tr>
<td>Sensitivity ASM</td>
<td>2</td>
<td>12</td>
<td>5</td>
<td>14</td>
<td>4</td>
<td>13</td>
<td>3</td>
<td>11</td>
<td>10</td>
<td>9</td>
<td>7</td>
<td>1</td>
<td>8</td>
<td>6</td>
</tr>
</tbody>
</table>

● Identical ranking between the standard regression coefficients and the partial regression coefficients.

● Some differences between the regression methods and ASM method.

● Contribution of each parameter to the overall uncertainty

● Linear regression:

\[ R = \sum a_i X_i \]

parameter contribution: \[ \Delta R = a_i \Delta X_i \]

● ASM method:

\[ R = R(X_1, \ldots, X_p) \]

parameter contribution: \[ \Delta R = \int \frac{\partial R}{\partial X_i} \Delta X_i \]

<table>
<thead>
<tr>
<th>Parameters</th>
<th>D</th>
<th>C_{mw}</th>
<th>\lambda_w</th>
<th>H_{cl}</th>
<th>H_{nb}</th>
<th>H_{co}</th>
<th>T_{tr}</th>
<th>Q_{le}</th>
<th>Q_{ve}</th>
<th>C_{l}</th>
<th>C_{g}</th>
<th>T_{obb}</th>
<th>T_{oa}</th>
<th>E</th>
</tr>
</thead>
<tbody>
<tr>
<td>regression</td>
<td>153</td>
<td>0</td>
<td>18</td>
<td>36</td>
<td>127</td>
<td>15</td>
<td>50</td>
<td>96</td>
<td>2</td>
<td>75</td>
<td>42</td>
<td>1847</td>
<td>16</td>
<td>10</td>
</tr>
<tr>
<td>( \Delta R ) in g.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>ASM</td>
<td>220</td>
<td>9</td>
<td>47</td>
<td>0</td>
<td>134</td>
<td>1</td>
<td>176</td>
<td>12</td>
<td>16</td>
<td>16</td>
<td>34</td>
<td>1940</td>
<td>25</td>
<td>45</td>
</tr>
</tbody>
</table>

Comparable results for the important parameters
Purpose

effect of deletion of one parameter

Parameter Tobb excluded of the uncertain parameters

<table>
<thead>
<tr>
<th>Results</th>
<th>Reference study</th>
<th>Tobb excluded</th>
</tr>
</thead>
<tbody>
<tr>
<td>Experiment</td>
<td>[13.300, 14.700]</td>
<td>[13.300, 14.700]</td>
</tr>
<tr>
<td>Wilks formula</td>
<td>[13.224, 15.095]</td>
<td>[13.927, 14.155]</td>
</tr>
</tbody>
</table>

• Purpose

investigation of the effect of the sample on the confidence interval

<table>
<thead>
<tr>
<th>Sample</th>
<th>Wilks formula</th>
<th>normal distribution hypothesis</th>
<th>Monte Carlo</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>N'</td>
<td>conf interval</td>
<td>AM kg</td>
</tr>
</tbody>
</table>

The difference is:

2 % with the Wilks formula
5 % with the normal distribution hypothesis

The confidence interval depends slightly of the choosen sample, with the two methods
• Purpose

investigation of the effect of the sample on parameter ranking

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Linear multiple regression</th>
<th>A.S.M. method</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>sample 1</td>
<td>sample 2</td>
</tr>
<tr>
<td>D</td>
<td>-7.633</td>
<td>-10.201</td>
</tr>
<tr>
<td>λ</td>
<td>-0.002</td>
<td>0.334</td>
</tr>
<tr>
<td>C</td>
<td>+0.707</td>
<td>-3.175</td>
</tr>
<tr>
<td>H</td>
<td>-0.036</td>
<td>-0.020</td>
</tr>
<tr>
<td>H</td>
<td>-0.127</td>
<td>-0.120</td>
</tr>
<tr>
<td>H</td>
<td>-0.015</td>
<td>-0.013</td>
</tr>
<tr>
<td>T</td>
<td>-0.505</td>
<td>-0.056</td>
</tr>
<tr>
<td>Q</td>
<td>+0.096</td>
<td>-0.008</td>
</tr>
<tr>
<td>Q</td>
<td>-0.002</td>
<td>+0.032</td>
</tr>
<tr>
<td>C</td>
<td>+0.075</td>
<td>+0.081</td>
</tr>
<tr>
<td>C</td>
<td>-0.042</td>
<td>+0.027</td>
</tr>
<tr>
<td>Tobb</td>
<td>-1.847</td>
<td>-1.873</td>
</tr>
<tr>
<td>Toa</td>
<td>+0.016</td>
<td>+0.000</td>
</tr>
<tr>
<td>E</td>
<td>+0.110</td>
<td>+0.049</td>
</tr>
</tbody>
</table>

* values not comparable

The results found by the ASM Method depends slightly of the sample contrary to these given by the linear multiple regression.

• MAIN FEATURES OF THE METHODOLOGY:
  • rigorous mathematical tool
  • reduction of expert judgement
  • flexibility (any scenario)

• RESOURCES:
  • 2 to 6 men-year for the development
  • mastering of computer time

• FURTHER STEPS:
  • estimation of parameter uncertainties
  • investigation on scaling effect (counter part analysis)
  • interpretation and verification on test cases (Integral Tests)
• RECALL THE GENERAL PROBLEM:

- to find the standard deviation of all the relevant parameters
(potentially contributing on response uncertainty)
- rather easy for parameters which the experimental can be directly measured.
ex: \( \lambda_{\text{UO}_2} \), heat transfer coefficient.

• DIFFICULTY:

- there is a problem for parameters (described by a correlation) which cannot be directly measured.
ex: interfacial friction
   liquid-interface heat transfer

1- to list the correlations describing parameters \( \varepsilon_j \) not directly measured.
(\( \approx 25 \) parameters: estimation of the Cathare team)

2- to list the Separate Effect Tests (SETs) which are sensitive to the effect described by the correlations.
(\( \approx 15 \text{ exp.} \Rightarrow 200 \text{ tests.} \))

3- to choose a set of responses, sensitive to the chosen parameters (\( \approx 500 \) responses).

4- to calculate the standard deviation \( \sigma_R_j \) of these responses:
   - code-experiment differences,
   - experimental uncertainties.

5- to perform one CATHARE+ASM calculation per test to obtain the sensitivities of the responses to the parameters \( dR_j/d\varepsilon_j \).

6- from:
   - the standard deviations of responses \( \sigma_R_i \)
   - the sensitivities \( dR_j/d\varepsilon_j \).

get:
   - the standard deviation of parameters \( \sigma_{\varepsilon_j} \)
   - the covariance between them, \( \text{cov}(\varepsilon_j, \varepsilon_k) \).
**PARAMETER UNCERTAINTIES**

(CONT'D)

- **DISADVANTAGE:**
  
  Step 6 requires mathematic developments.
  (testing underway)

- **ADVANTAGES:**
  
  - calculation of the dependencies between parameters (covariances).
  - very general methodology (can be performed systematically during assessment).
  - low CPU cost.
  - limitation of the expert judgment for the choice of the set parameters-responses.

- **MATHEMATICAL CRITERIA HAVE BEEN FOUND:**
  
  - to eliminate not very influent parameters
  - to check if important parameters were forgotten

---

**THE PRESENT STATUS**

- one correlation: interfacial friction for bubbly-slug flows in pipe geometry (3 parameters).

- one experiment (Vertical Canon) used to establish this correlation (5 tests).

- 20 responses measured in Vertical Canon:
  
  - remaining mass of water
  - void fraction at ≠ elevations

- the mathematical developments needed in 6 were performed for this simple case.

  - result: $\tau_j$ known within a factor of $\approx 2$. 
Application of the methodology with the set of S.E.T. facilities

THE PROBLEM:

We have a set of:
- \( d \) parameters \( \xi_j \), representative of the closure relations
- S.E.T. facilities, with different responses \( R_i \)
  - in each facility, only one or several parameters are important
  - the same parameter can be important in several facilities.

the aim:

- to find the covariance matrix of the \( d \) parameters

THE METHOD:

- for each facility:
  to calculate the sensitivities of the responses to all the parameters:
    \[
    \frac{\partial R_i}{\partial \xi_j} \quad j = 1, d
    \]

- for all the responses of all the facilities:
  only one calculation:
  \[
  \frac{\partial R_i}{\partial \xi_j} \quad j = 1, d
  \]

- all the sensitivities

- the standard deviations of the responses: \( \sigma_{R_i} \)

\[
\begin{align*}
\text{one covariance matrix} & \quad \text{for all the parameters} \\
\end{align*}
\]

- use of a mathematical criterion to eliminate the parameters, which are the least important for the set of facilities
1. \( \text{facility 1: } E_1 \text{ important, } E_2: \text{ not very important} \)
   \[ \text{facility 2: } E_2 \text{ not very important, } E_1: \text{ important} \]

   - to treat both facilities together

   - to check that:

   \[
   \begin{align*}
   \text{facility 1} & \rightarrow \text{var } E_2 : \text{ not perturbed by facility 2} \\
   \text{facility 2} & \rightarrow \text{var } E_1 \\
   \text{var } E_1 & = 1
   \end{align*}
   \]

   if problems:

   - to consider only the important parameters
     for each facility

   - the risk: to forget some covariances

2. The same parameter \( E \), important in 2 facilities.

   BUT:

   \[
   \begin{align*}
   \text{E well fitted for facility 1} \\
   \text{E badly } & \sim \sim \sim \sim \sim \sim \sim 2
   \end{align*}
   \]

   if same weight for each facility:

   \[
   \tau_1^2 = \tau_1^2 (\text{facility 1}) + \tau_2^2 (\text{facility 2})
   \]

   problems to solve:

   - weight for each response of each facility?

   - number of responses per facility?
1. INTRODUCTION

In the past, EVALUATION type codes have been used exclusively by the NSSS suppliers for plant licensing. This approach was dictated by the absence of rigorous physical models to deal with complex plant phenomena, and by the concern to provide a conservative EVALUATION of the plant behaviour for a particular phenomenon (e.g. DN3, LOCA PCT). The drawback of these codes is that they concentrate on a few critical phenomena and can hide the real process.

Following 2 decades of intense research in the area of THERMAL-HYDRAULICS, the so called "BEST ESTIMATE" codes have been developed and are being assessed on a wide variety of experimental data from both small scaled facilities. These codes are now available for performing a realistic and physically based thermal-hydraulic analysis of the reactor systems behaviour, and also for licensing.

Indeed, in 1988, the USNRC revised the emergency core cooling system (ECCS) licensing rule to allow best estimate computer codes for calculation of the ECCS behaviour, provided an uncertainty quantification is performed (Regulatory guide 1.157). This change in the regulatory practice opens the potential to use these advanced best estimate codes also for a large number of licensing calculations, provided an acceptable methodology is developed and accepted by the safety authorities for each application.

Chapter 2 presents the Belgium context and the fundamentals of the BELGATOM methodology on the basis of which a practical application is applied and discussed in chapter 3.

2. BELGATOM METHODOLOGY FOR PLANT LICENSING ON THE BASIS OF THE CODE RELAP-5/MOD2

2.1. LICENSING BACKGROUND.

In BELGIUM, a total of 7 nuclear power plants of the PWR type (with a total capacity of 5450 MWe) are installed, providing 60% of the total electricity production.

Since several plants experience problems from corrosion defects in the steam generators (â"SGâ), a program of SG replacement is scheduled for at least 3 power plants: DOEL3 (1993), TINANGE 1 (1995) and DOEL 4 (1996).

To recover the large costs from such major hardware change, uprating the power plant is an attractive economic possibility.

On the one hand, the feasibility to perform a plant uprating depends on the hardware characteristics such as:

- The power margin from the turbogenerator which for all plants in BELGIUM is substantial.
- Advanced fuel concepts which can offer improved core performance,
- The improved heat transfer characteristics and performance of the replacement SGs.
- The design margins of the active safeguard systems.

On the other hand, from the licensing point of view, a power uprating is determined by applying improved thermal design procedures for the core and advanced best estimate system codes for evaluating the performance of the active safeguard systems and for identifying the real safety margins of the plant for various design basis accidents.
These features enabled a power uprating of DOEL-3 NPP (a 900 MWe 3 loop plant) by 10%. BELGATOM performed NON-LOCA calculations by means of the code RELAP-5/MOD-2 for which an acceptable methodology was elaborated at the request of the Belgian Safety Authorities.

2.2. BASIC PRINCIPLES OF THE BELGATOM LICENSING METHODOLOGY.

The methodology seeks to provide a bounded analysis of the expected plant behaviour under the postulated accident by means of best estimate thermal-hydraulics codes RELAP-5 and COBRA. The accidents covered by this methodology exclude:

* all reactivity accidents for which a point kinetics model is not acceptable such as a rod ejection, rod drop, streamline breaks, etc.
* all LOCA's for which a quantification of the margins is requested if a best estimate code is used.

The methodology is based upon: (fig 1)

a. The use of the best estimate advanced thermal-hydraulics code RELAP-5/MOD-2
b. The application of a detailed standard nodalisation of the plant based on the code user guidelines and assessed on the basis of experimental evidence from various small scale integral test facilities and full scale plant transient recordings.

c. The selection of an acceptable key licensing parameter for each type of safety analysis, and the quantification of its acceptable limit, based on the legal licensing criteria.

d. The application of a limit value approach by which all plant initial and boundary conditions as well as the protection system tolerances are assumed at their most adverse conditions for the selected key parameter.

e. The evaluation of a code bias for the selected key parameter, resulting from known code weaknesses and code uncertainties.

3. APPLICATION : FEEDWATER LINE BREAK ACCIDENT.

While the methodology is generic, the following plant specific results are presented for illustrative purposes only and are obtained for the DOEL-3 power plant equipped with new SG's.

![Diagram](image)

3.1. GENERAL DESCRIPTION OF THE ACCIDENT PHENOMENA.

The feedwater line break is a class 4 accident which constitutes one of the limiting accidents for sizing the auxiliary feedwater system.

The scenario for this accident, illustrated in the figures 2 to 6, assumes the break to occur at t=30 s downstream of the feedwater isolation valves in the main feedwater pipe. The plant loops are labelled respectively R, G and B, while the break occurs on the SG of loop B, which also contains the pressuriser.

The double ended break precludes further water addition to the intact SG's, while the affected SG loses inventory through the break.

Due to depletion of the water inventory of the intact SG's, the water level drops, until the low-low SG level is reached in the intact SG's, which triggers reactor trip and turbine trip at 72.0 s. Steamline isolation occurs in all main steamlines 5 s after reactor trip which effectively isolates the intact SG's from the broken SG and stops any reverse blowdown from the intact SG's. While the intact SG pressure increases up to the setpoint of the safety valves, the affected SG depressurises completely down to atmospheric pressure at about 150 s.

The auxiliary feedwater pumps start feeding water to the 3 SG's 60 s after reactor trip and begin delivering 40 t/hr to each intact SG at 132.0 s.

The long term figures (0 to 2400 s) illustrate respectively:

* The loop average temperature (Fig. 2),
3. The void fraction in the core (Fig. 3).
4. The reactor coolant system (RCS) subcooling (Fig. 4).
5. The reactor vessel collapsed water level (Fig. 5).

Figure 5 illustrates that the large volumetric expansion resulting from bulk boiling, causes an enhanced depletion rate of the RCS inventory via the relief valves, and core collapsed levels start to decreases until the total auxiliary feedwater flow is realigned to the intact SG’s, 30 minutes after reactor trip.

The main thermal-hydraulic phenomena observed during such accident and for which the code must be assessed can be summarised as follows:

- Heat transfer degradation in the SG’s,
- Break flow in a complex geometry,
- Residual heat removal by the intact SG’s and structural heat storage,
- Primary system heat-up, pressurisation and voiding,
- Two phase flow behaviour in the SG’s and the RCS.

This accident when not properly controlled could lead to a primary overpressurisation and/or a significant core heatup due to core recovery and possible risk of insufficient core cooling.

3.2. ACCEPTANCE CRITERIA FOR A FEEDWATER LINE BREAK ACCIDENT.

While the Standard Review Plan lists the specific criteria to be used in evaluating the consequences of the feedwater line break, the more limiting criteria that will be adopted for the purpose of evaluating the consequences of such accident are as follows:

1. The pressure in the reactor coolant and main system should be maintained below 110 percent of the design pressure.
2. The liquid mass in the RCS will be sufficient to cover the reactor core at all times thereby ensuring sufficient core cooling.

For the primary pressurisation criterion, this accident is normally less limiting than the accident resulting from a sudden loss of load (e.g. closure of all MSIV’s) which is the design basis accident for sizing the pressurizer relief and safety valves.

The most limiting criterion for a feedwater line break is the verification that the core remains covered with water at all times during the accident. Hence for the remainder of this study the key licensing parameter is the reactor vessel collapsed water level and the key criterion is: "no core uncoverage".

3.3. RELAP-3/MOD-2 CODE ASSESSMENT BASIS AND APPLICABILITY.

Within the framework of ICAP (International Code Assessment Program), numerous assessment studies were performed by the members from a large number of international organisations, of which 62 studies were selected to identify recurrent code deficiencies to be addressed in future code improvements.

Nine major code deficiencies (including various facets of each deficiency) were identified.

From the overall description of the important thermal-hydraulic phenomena in chapter 3.1, one can identify 2 deficiencies which may have an adverse impact on the key parameter:

- Interfacial friction in bubbly/slug flow-regime,
- Critical flow at the break.

3.4. PLANT PARAMETRIC AND SENSITIVITY STUDIES.

3.4.1. Determination of the REFERENCE calculation.

The so called reference calculation is based on the limit value approach; this means that any other initial and boundary conditions, taken within the definable range of values, would result in an improved margin on the key parameter with respect to the acceptance criterion.

Therefore, the first activity concerns the identification of those plant parameters which may have an impact on the evolution of the feedwater line break accident.

Next, the sense of the deviation of each plant parameter leading to a negative impact on the key parameter is determined by engineering judgement and/or by performing calculations if required.

It must be emphasised that engineering judgement alone can be misleading if two counteracting processes or nonlinearities may be involved. Indeed, the variation of a parameter (e.g. the break size) may also affect the timing of the reactor protection systems (e.g. reactor trip) such that the impact on the key parameter is a priori uncertain.

3.4.2. Sensitivity studies.

The sensitivity studies are performed next by resetting the conservative initial and boundary conditions of the reference calculation, to the best estimate value, one at a time.

The objectives of these sensitivity studies are threefold:

- to confirm that the options taken for the reference calculation indeed combine the most adverse plant conditions in terms of the key parameter,
• to identify the parameters for which a parametric study is required for each plant licensing study since the impact of plant specific systems can change the direction of the conservatism,
• to quantify the impact of each individual parameter variation on the key parameter. The parameter change which yields with the largest impact on the key licensing parameter, constitutes the single failure criterion.

Sensitivity studies for a feedwater line break were performed for each parameter, of which some parameters of special importance for such accident were examined such as:

• Impact of the initial SG water inventory,
• Impact of the structural mass,
• Impact of non perfect mixing in the reactor core inlet and outlet, since this accident must be considered as a non-symmetric accident,
• Impact of the reactor coolant pump trip...

The results of all the sensitivity studies are summarised in figure 6.

3.5. QUANTIFICATION OF THE CODE BIAS.

To evaluate the impact of the code weaknesses, several calculations are performed using the reference calculation as the starting point for each code parameter study. The code bias (not in the statistical sense) constitutes the summation of all adverse quantities resulting from the various studies. This bias is added to the results of the reference calculation to arrive at the calculated value for the key licensing parameter.

From the identification of the important thermal hydraulic phenomena that control this accident evolution and the code deficiencies as well as code uncertainties which may have an impact on the criterion (chapter 3.3), 4 sensitivity studies, described in following paragraphs, are identified to quantify the code bias.

3.5.1. Impact of the water entrainment during the SG blowdown.

This effect is covered by selecting the no-slip option in the downcomer junctions in the affected SG such that maximum water entrainment occurs during blowdown. This option is used in the reference calculation and all subsequent sensitivity calculations.

3.5.2. Impact of the critical flow uncertainty and break size

In order to cover both the break size and the possible non conservative code behaviour for the critical flowsite, four additional runs are made to evaluate the impact of various break sizes respectively 130%, 75%, 50% and 25%. Due to compensating effects of earlier reactor trip and smaller remaining SG's inventory after trip for larger break sizes, there is no monotone relation between the break size and the collapsed water level. Since these results depend on the feedwaterline geometry and the protection setpoint, the full break spectrum must be analysed for each licensing calculation.

3.5.3. Impact of the uncertainty in the SG's heat transfer.

To account for the uncertainty in the heat transfer correlation's, 2 analyses were made with the boundary and initial conditions of the reference calculation but with varying the SG heat transfer area in all SG's from -25% to +30%.

3.5.4. Impact of the time step size.

Sensitivity studies were performed for requested time step sizes varying between 100 and 10ms (the convective Courant limit for this nodalisation was 110ms). While the number of repeated advances reduces for smaller requested time step size, the number of cycles increases almost linearly with the inverse of the step size as well as the mass reduction due to truncation errors.

Comparison of the SG mass error and the minimum collapsed water level in the RCS for different requested time step sizes revealed a striking correlation, indicating that the mass loss in the SG's by numerical truncation is equivalent to reducing the effective auxiliary feedwater to the SG's. This observation was confirmed by a sensitivity study at the reference requested time step size (100 ms) but a reduced auxiliary feedwater flow equivalent to calculated SG mass reduction for a 10 ms requested time step size.

From this studies it can be concluded that any negative mass error in the SG's constitutes an inherent conservatism for the evaluation of the key parameter and that no additional bias should be added to account for the impact of the time step size.

3.5.5. Determination of the maximum code bias.

Figure 6 illustrates also the impact of the different parametric studies for the code uncertainties.

The summation of all negative code biases only, results in a penalty, to be subtracted from the reference calculation which then gives the licensing value for the key parameter.

4. CONCLUSIONS.

• The methodology, based on the use of a best estimate thermal-hydraulic code, applicable to most non-LOCA accidents, presents a practical and licensable approach to identify and quantify in a systematic way the impact of:
  - all adverse plant initial and boundary conditions,
- the code deficiencies and uncertainties.

- The benefit of using best estimate codes is obtained by providing
  - a better insight in the real problems
  - additional margins for power uprating which would otherwise be impossible or have to be covered by substantial hardware changes.

- This method decouples completely the uncertainties in the plant initial and boundary conditions from the code uncertainties. Hence, any improvement of the code, can readily be assessed starting from identical "reference" conditions.

- Since the plant conditions and code uncertainties are not treated in a statistical way, and require therefore much less effort, there still remains an unquantifiable inherent conservatism in the results.

- Engineering judgement alone can be misleading since the expected direct consequence of a parameter change can be completely offset by other non-linear effects such as the impact on the timing of the reactor trip or valve opening history.

REFERENCES:


2. M. STIEVENART, W. LODEWIJCKX and J. ROSEYS, BELGATOM. "Optimisation of the operating conditions of pressurised water reactors limited by the DNB phenomenon." NURETH 6, October 1993, Grenoble FRANCE.


Figure 6 : Summary of results for Deel 3
Limit Value Approach (LVA) for Uncertainty Evaluation of Best Estimate LOCA Analyses

Naugab E. Lee   Gerhart Menzel

ABB-CE Nuclear Operations
Combustion Engineering, Inc.
Windsor, Connecticut 06095

ABSTRACT

ABB Combustion Engineering has developed the Limit Value Approach (LVA) as a unique and conservative approach to address the uncertainty issue of the revised ECCS rule. The LVA simplifies the uncertainty analysis which might be necessary if the NRC's CSAU methodology were explicitly followed.

The justification of the LVA methodology is provided by evaluating its underlying assumption. A comprehensive statistical uncertainty analysis verifies that the LVA-based PCT is higher than the statistically evaluated PCT leading to a conclusion that the LVA is a conservative approach.

The reason behind developing the LVA methodology is to provide a simple economical means for licensing LOCA analyses.

Introduction

The revised Emergency Core Cooling System (ECCS) rule (Reference 1) allows the use of a best estimate computer code for a loss of coolant accident (LOCA) analysis if the uncertainty of the calculations is quantified and accounted for. This rule became effective on September 16, 1988. A detailed guideline for using the revised licensing rule was published in the Spring of 1989 in a Regulatory Guide (Reference 2).
The US NRC has developed an uncertainty evaluation methodology to demonstrate that a code uncertainty may be quantitatively evaluated under the new licensing rule for a large break LOCA analysis. This methodology is called Code Scaling, Applicability and Uncertainty (CSAU) methodology (Reference 3). The NRC applied the same methodology to small break LOCA transients for a Babcock & Wilcox PWR (Reference 4).

Limit Value Approach

The limit value approach (LVA) at ABB Combustion Engineering is defined as "analyses in which licensing peak clad temperatures (PCTs) are evaluated with a best-estimate code by setting key plant parameters at their limit values." Reference 5 describes the initial development of the LVA methodology and is attached to this paper as an appendix.

This approach needs the following components:

Generic
1. Best-estimate code and its input model
2. Set of key plant parameters (that substantially affect PCT)

Plant Specific
1. Limit values (that maximize PCT) of key plant parameters

ABB Combustion Engineering is utilizing CEFLASH-4AS/REM and PARCH/REM as the best estimate codes for small break LOCA analyses. (Reference 6) We, together with EPRI, have also utilized RELAP5/MOD2 to study the feasibility of using the LVA methodology for SBLOCA transients. (Reference 7) Currently ABB Combustion Engineering utilizes RELAP5/MOD3 as the best estimate code for large break LOCA licensing analysis.

The selection of the key parameters is based on expert opinion/experience and two-level fractional factorial design analyses. (Reference 8) In a two-level factorial design, an analyst
selects two levels for each of a number of variables and then examines all possible combinations of the levels. A fractional factorial design investigates a limited set of the all possible combinations. This approach is more effective in finding major effects than one-at-a-time investigations. A factorial design also addresses parameter interaction effects.

A two-level factorial design is especially useful in accounting for not only the physical importance of the parameters but also their uncertainties. It is important to account for the parameter uncertainties in selecting key parameters because the overall calculational uncertainty is a function of both their impact and uncertainties.

Considering the impact only is all right when we consider a deterministic result. However, when we consider uncertainties, this approach is not sufficient because of the reason explained below. This is illustrated by the following simple example:

Let a system can be described as the following general function:

\[ y = f(x_1, x_2, \ldots, x_n) \]  \hspace{1cm} (1)

We may expand the function with a Taylor series and then ignore second- or higher-order terms assuming a linearity. Then the variance of \( y \) may be given as the following equation assuming all covariance terms are zero:

\[ \sigma_y^2 = \sum_{i=1}^{n} (\frac{\partial f}{\partial x_i}|_{x_i})^2 \sigma_{x_i}^2 \]  \hspace{1cm} (2)

where the partial differential terms are evaluated at the mean \( x \) values. This equation shows that the final variance of the result of the equation is the sum of the squares of the products of the component derivatives and their uncertainties. Certainly, a PWR LOCA transient uncertainty
is not calculated by the above simple relation. However, the message we get from this discussion is still very true.

Plant specific limit values for the key parameters should be determined from each plant condition.

Justification of LVA

To utilize the LVA methodology, we need to justify it by showing that the underlying assumption of the LVA is sound and the LVA-based PCT is an upper bound of the statistically evaluated PCT.

The underlying assumption of the LVA is that the overall system response (PCT) will be maximized if each of the key parameters is set to maximize its individual effect on the system. A condition to satisfy this assumption has been developed mathematically and we were able to show that a SBLOCA analysis meets the condition.

A comprehensive statistical uncertainty evaluation, which is similar to CSAU, has been performed for SBLOCA to compare an LVA-derived PCT to a statistically evaluated PCT. The next section describes the statistical uncertainty evaluation method.

Statistical Evaluation of PCT

Even though a code calculates a single PCT with a given set of inputs, it is conceivable that the temperature will have a bounded distribution if the inputs are changed to reflect uncertainties from various sources. To represent this situation, a code calculated PCT may be presented as

\[ T = f_{\text{code}}(NC, AU, BU, CU, DU, EU, FU) \]  

(3)
where

NC: Nominal conditions,

AU: Uncertainty from simplifying assumptions and approximations in models like three-dimensional effects and annulus bypass flow,

BU: Uncertainty in boundary and initial conditions such as power level, control systems and available equipment and performances,

CU: Code uncertainty

DU: Data uncertainty and applicability

EU: Engineering uncertainty

FU: Fuel uncertainty such as fuel conductivity, gap width, gap conductivity, decay heat and peaking factors.

Equation 3 may be conveniently reformulated as follows to take out the uncertainty portion from the nominal condition result:

\[ T = T_{NC} + g(AU, BU, CU, DU, EU, FU) \]  \hspace{1cm} (4)

The values for the function \( g \) in Equation 4 will have a limited range for a given level of confidence. This fact is pictorially presented in Figure 1. Note that the range is not necessarily symmetry about the best estimate temperature. The maximum positive value of \( g \) (maximum positive overall calculational uncertainty, \( \Delta T_{oc} \)) may be expressed as

\[ \Delta T_{oc} = g(AU^*, BU^*, CU^*, DU^*, EU^*, FU^*) \]  \hspace{1cm} (5)

where the superscript * means the set which cause the maximum g value. Then the licensing temperature may be obtained as
\[ T_{LC} = T_{NC} + \Delta T_{OC} \] 

(6)

If we assume that the component uncertainties are independent and their impacts are additive, then Equation 5 may be recast as

\[ \Delta T_{OC} = \Delta T_{code}(CU^*, DU^*) + \Delta T_{pf}(BU^*, FU^*) + \Delta T_{p}(AU^*, EU^*) \] 

(7)

The first term of Equation 7 is the code uncertainty. This term will be evaluated based on the component models through simulations of separate effect and integral effect tests. This will account for data uncertainty and data applicability. The second term is related to the uncertainty of initial and boundary conditions and fuel conditions. These uncertainty sources are minimally related to the code predictability. The third term represents uncertainties which may not be quantified as distributions so that they may be represented as a bias.

Equations 6 and 7 can form a basis for a licensing temperature calculation. Each of the terms on the right hand side of Equation 7 is explained in more detail in the following sections showing how they can be evaluated.

A. **Code Uncertainty**

This section discusses how the first term on the right hand side of Equation 3.3-7 can be evaluated. This code uncertainty term is dependent on both code calculational uncertainty and the data applicability to the transients of interest. Code calculational uncertainty measures the uncertainty related to thermal hydraulic responses. Instead of analyzing an uncertainty caused by each component and model of a code utilized in LOCA calculations, the present approach looks at code capabilities to calculate a key parameter (for example, PCT). This means that the methodology concentrates on the combined effect of component models on a key parameter for
a specific application. Because of its attention to a particular application, this approach may be called Application Based Code Uncertainty Evaluation (ABCUE).

If initial and boundary conditions which drive the key parameter transient were known, separate effect test simulation with the boundary conditions will tell exactly what the code bias and uncertainty are in calculating the key parameter transient. Therefore, this approach to estimate a code uncertainty should be acceptable if the following three conditions are met:

1. A code should be able to calculate a transient reasonably well catching important phenomena and their timing of the occurrence.

2. There should be enough separate effect test data to cover the boundary conditions so that statistical analyses become meaningful in terms of the data variance.

3. No scaling distortions are involved.

Each of these conditions must be considered carefully to evaluate this approach of calculating a code uncertainty. To address the first condition, integral effect tests can be used to test the code predictability. If integral test predictions are within the "code uncertainty" estimated from separate effect test (SET) simulations, it may be concluded that the code is generally predicting the transients of interest very well and we can use the SET-based code uncertainty as the inherent code uncertainty for the key parameter calculation. If the IET predictions are outside of the key parameter uncertainty, then the code predictability must be improved. This code improvement effort should be in the area of individual models and correlations.

The second item has to be judged based on available data bases and the test matrix must be selected based on possible boundary condition transients for the key parameter. Naturally, it calls for as much data as possible.

The third item is on the scaling impact on the data. Since this methodology is trying to simplify uncertainty evaluation by concentrating on a key parameter of an application, it is highly desirable to have SET data with minimum scaling distortions.
This methodology concentrates its attention on a key parameter of application for the evaluation methodology. For the LOCA application, the key parameter is PCT. Code predictability of the system transient response is evaluated against IET tests. This type of simpler thinking is very attractive in approaching a phenomenon for which we do not have all relevant information to render a reasonably complete statistical analysis.

The data applicability is related to scaling uncertainty. For this approach, since the PCT-related separate-effect tests data may be selected with minimum scaling distortions, this uncertainty may be very small.

Some scaling effect (such as three-dimensional effect) may be accounted for as a part of the bias term ($\Delta T_b$) which will be discussed later.

The impact of uncertainty propagation during the transients is already considered if the separate effect tests are selected to cover wide range of conditions.

The code uncertainty may be different depending on its application. Therefore, the code uncertainty has to be considered with a specific application in mind. For our application to LBLOCA analyses, this must be the code uncertainty for 'calculating PCT during a LBLOCA at a PWR.'

B. **Plant Parameter Uncertainty**

This section covers the second term in Equation 7. This term is related to uncertainties on plant initial and boundary conditions and fuel behavior.

These conditions can be modeled in a best-estimate manner and their associated uncertainties may be accounted for. However, some of these conditions may affect the resulting calculations significantly enough so that their impacts are better treated as separate sets of data than mixing them together. This suggestion is similar to the blocking technique in a statistical analysis.
To make the matter simpler, we need to find a set of conditions which will cause a worst transient in terms of PCT. Then statistical combination of their uncertainty through a response surface gives the magnitude of this uncertainty within a selected confidence level.

C. Bias (Application Uncertainty)

The last term in the right hand side of Equation 3.3-7 is uncertainties related to parameters and conditions which we do not have much control over or we do not have a good understanding. Therefore, this term is better treated as a bias which may be called an application uncertainty. Three-dimensional effects in the vessel, annulus bypass, etc. may be accounted for here.

The approach for estimating this bias depends on data availability. For the cases where we have dependable data, the biases can be directly obtained by comparing the data and calculation. For the effects where we do not have reasonable data, some parametric calculation for the effect will help establish the bias.

The Limit Value Approach (LVA) accounts for plant parameter uncertainties in the calculation of PCT. Each LVA calculation simultaneously applies the maximum adverse uncertainties of the significant plant parameters. The magnitude of the PCT differential between calculations with nominal and with Limit Values varies with the scenario and with the plant parameters assumed in the calculations.

Summary and Conclusions

This paper briefly describes the LVA methodology which is being used at ABB-CE to implement the new revised ECCS rule in LOCA analyses. This work has shown that the LVA methodology meets the new ECCS rule requirements.
is intended for NRC required analyses of breaksize spectra.
- is simple and fast so that it is economical
- is conservative relative to statistical methodology

In addition, a statistical methodology to evaluate the analysis uncertainty has been developed in the course of verifying the LVA methodology. This statistical methodology may be utilized if an additional margin is required.

References


6. Small Break LOCA Realistic Evaluation Model. CEN 420-P (Proprietary), October,

Schematic Relation for Licensable PCT and LVA PCT

\[ T_{NC} = \text{PCT at Nominal Conditions} \]
\[ T_U = \text{PCT at Upper } 2\sigma \]
\[ T_{LVA} = \text{PCT Calculated with LVA} \]

Figure 1. Schematic View of Temperature Relations
APPENDIX

Implementation of Best Estimate Analysis in PWRs - One Vendor’s Approach

G. Menzel
Combustion Engineering, Inc.

Presented at EPRI Workshop on Appendix K Relief Using Best Estimate Methods:
The Revised LOCA/ECCS Rule
August 11-12, 1988; Cambridge, MA.

ABSTRACT

In order to benefit from the proposed LOCA/ECCS Rule amendment, Combustion Engineering has developed a realistic small break LOCA evaluation model and an uncertainty estimation technique. The benefits from this evaluation model will be used by utilities to ease plant equipment surveillance requirements or plant operating limits. With respect to the uncertainty estimation of the analysis results required by the proposed Rule amendment, Combustion Engineering’s Limit Value Approach (LVA) conservatively accounts for the uncertainty due to plant conditions and parameters. Uncertainties in computer programs and calculational models are estimated by comparisons of analyses to appropriate test data. Realistic small break LOCA calculations using the LVA together with estimates for computer program and model related uncertainties have shown significant reductions in cladding temperature relative to the current Appendix K calculations. Combustion Engineering together with EPRI has outlined a project to implement the LVA for realistic small and large break LOCA calculations using the RELAPS/MOD2 computer program.
INTRODUCTION

At present, for licensing of nuclear power plants the performance of the emergency core cooling system (ECCS) during a loss-of-coolant-accident (LOCA) is analyzed according to rules described in the Code of Federal Regulations, 10CFR50.46 and its accompanying Appendix K. The proposed amendment to these rules will allow the use of best estimate or realistic calculational models together with an estimation of the uncertainty of the calculated results. The purpose of the amendment is to provide an alternative to the present LOCA analysis rules which will permit removal of unnecessary conservatism from the LOCA calculations. Thus, in order to benefit from the LOCA/ECCS Rule amendment, realistic calculational models and methods to estimate the uncertainties need to be available.

UTILIZATION OF BENEFITS FROM AMENDED LOCA/ECCS RULE

Several utilities with Combustion Engineering nuclear power plants are interested in utilizing the benefit from realistic LOCA calculations to ease surveillance requirements for plant equipment or operating limits. An example of equipment surveillance requirements is the testing of the high pressure safety injection pump performance, while axial power shape index is an example of an operating limit. Currently, the surveillance requirements or operating restrictions which utilities are interested in relaxing are related to the small break LOCA calculations. Therefore, Combustion Engineering has developed a realistic small break LOCA evaluation model and an associated uncertainty estimation method.

REALISTIC SMALL BREAK LOCA EVALUATION MODEL

The Combustion Engineering realistic small break LOCA evaluation model comprises the calculational models in the thermal-hydraulic system response computer program CEFLASH-4AS and in the hot rod heat-up computer program PARCH. Substantial modifications have been made to existing versions of these programs in order to come up with a physically realistic small break LOCA
evaluation model. These modifications involve global changes to the programs as well as changes to specific component models. Examples of global changes are a five equation thermal non-equilibrium model, a detailed phase separation model and a detailed wall heat transfer model. Examples of component model changes are: improved core heat transfer model, more realistic break flow model, improved steam generator heat transfer model, inclusion of counter-current flow limits in steam generator tubes and hot legs and improved loop seal clearing model. While the individual component model changes, in general, have been derived from separate effects tests, the entire realistic small break evaluation model has been successfully compared against small break LOCA tests at integral test facilities like Semscale, LOFT and ROSA IV/LSTF. The realistic small break LOCA evaluation model (1) has been submitted to NRC for approval.

EVALUATION OF UNCERTAINTIES

The proposed LOCA/ECCS Rule amendment does not specifically prescribe uncertainty evaluation techniques. Instead, it is stated that in the past the NRC has found a method acceptable that was judged to be at least at the 95% probability level. A general methodology for determining the uncertainties in LOCA calculations is described in Chapter 4 of the NRC report NUREG 1230 (2) and is referred to as "Code Scaling, Applicability and Uncertainty Evaluation Methodology" or CSAU. In CSAU, the total uncertainty is divided into the following four components which account for uncertainties due to:

1) Codes and Experiments
2) Scaling
3) Code Deficiencies
4) Plant Conditions and Parameters

These uncertainty components are added in CSAU to the realistic LOCA analysis results, e.g. peak cladding temperature (PCT), which were calculated with realistic (e.g. nominal) plant conditions as input and with realistic calculational models. This approach is graphically shown in Figure 1 under "CSAU (NRC) Approach".

-3-
LIMIT VALUE APPROACH

The Limit Value Approach (LVA) addresses the estimation of the uncertainty component due to plant conditions and parameters. In LVA, plant conditions and parameters which significantly affect the peak cladding temperature are set simultaneously at their Limit Value. For example, in the versions of the C-E computer codes CEFLASH-4AS and PARCH which are used for the realistic small break LOCA calculations about 20 of the input parameters which describe plant conditions or parameters are set at their Limit Value.

The selection of these significant parameters is straight-forward in some cases, for instance when selecting parameters describing decay heat generation or high pressure injection flow for small break LOCA calculations. The selection of other parameters requires experience with LOCA calculations in general and an understanding of the calculational models in the computer programs in particular. In the above-mentioned small break LOCA calculations with CEFLASH-4AS and PARCH, the significant parameters which were selected describe the safety injection system, core power and flow, core power shape, trip setpoints and steam generator secondary side conditions.

The Limit Value itself is that numerical value within the range of a parameter which would maximize PCT. Many of these parameters are measured or predicted values which have a degree of uncertainty associated with them. The determination of the specific value of a significant parameter which should be used as the Limit Value is, again, straight-forward in the cases where the effect on PCT of a particular parameter is obvious. For other significant parameters, the determination of the appropriate Limit Value may require the execution of specific sensitivity studies in addition to utilizing existing experience in LOCA calculations.

Calculations performed with the Limit Values yield a PCT which will be referred to in the rest of this paper as the Limit Value PCT. As mentioned before, the LOCA calculations are performed with realistic calculational models but with the Limit Value input. How this approach compares with the CSAU approach is schematically shown in Figure 1, under "Limit Value (C-E) Approach". It can be seen that with LVA the uncertainty due to plant
conditions and parameters is included in the calculation of the Limit Value PCT. In the LVA, the significant parameters are simultaneously set at their Limit Values in a deterministic manner in order to maximize PCT. Therefore, LVA is a more conservative approach than a statistical determination of a 95% probability level for the uncertainty in PCT due to plant conditions and parameters. LVA is also less expensive than the statistical approach, because for a given break size only one calculation is performed.

The Limit Value Approach has been discussed with the NRC Regulatory and Research staff and they did not raise any fundamental objections about the Limit Value Approach.

APPLICATION OF AMENDED LOCA/ECCS RULE

The application of the amended LOCA/ECCS Rule results in a significant reduction in PCT for small break LOCAs relative to the current Appendix K calculations. This is primarily due to the use of realistic calculational models which will be permitted by the amended LOCA/ECCS Rule. The PCT reduction is still substantial even when the uncertainties are included. The PCT for the "worst" small break (the break with the highest PCT) of a typical C-E NSSS of the 3410 MWₑ class is about 1730°F with the current Appendix K Evaluation Model. This PCT occurs during the boil-off phase of the small break when the core becomes temporarily uncovered by coolant before flow from the high pressure safety injection pumps recovers the core.

With realistic calculational models and including the LVA, the highest Limit Value PCT (without code-related uncertainties added) for a spectrum of small breaks ranging from about 2 in. to about 10 in. diameter is about 1150°F. Even if a value of 150°F for all code-related uncertainties is added, the resulting PCT of 1300°F is considerably lower than the Appendix K value of 1730°F. The break spectrum shows two ranges of high PCT. For breaks up to about 4 in. diameter, the PCT occurs during the boil-off phase when the core is uncovered for some length of time. Breaks greater than about 7 in. diameter also have high PCTs. For these breaks the primary system pressure drops low enough so that the safety injection tanks discharge and their flow together.
with the flow from the high pressure safety injection pumps becomes large enough to prevent core uncover. Instead, PCT occurs very early in the transient (within about 20 seconds after the break) during a very brief temperature spike of 10 to 20 seconds duration which is the consequence of a short film boiling period after Critical Heat Flux (CHF) is exceeded.

If the calculation described above is repeated with nominal values instead of Limit Values, the core does not uncover nor does it exceed CHF. The Nominal Value PCTs are about 700°F and are the steady state temperatures at the start of the accident. A comparison of these Nominal PCTs with the Limit Value PCTs shows that the magnitude of the PCT uncertainty related to plant parameters depends on whether with the Limit Value plant parameters the core experiences a sizeable core uncover or exceeds CHF. If core uncover or CHF occurs, the difference between Limit Value PCTs and Nominal Value PCTs is large, up to about 450°F. This indicates that for break sizes, which experience core uncover or CHF, the use of LVA is equivalent (in terms of CSAU) to adding an uncertainty of about 450°F for plant conditions and parameters to a cladding temperature which was calculated with nominal plant input parameters. For break sizes which experience with LVA neither core uncover nor CHF, the Limit Value PCTs become smaller and come close to the Nominal PCTs. Consequently, the difference between Limit Value PCTs and Nominal Value PCTs becomes small.

The PCT margin gain from the amended LOCA/ECCS Rule can be utilized in various ways. One choice is to use the margin to ease the surveillance requirements for the high pressure safety injection pumps by showing that acceptable PCTs are calculated even if the high pressure safety injection (HPSI) delivery flow were reduced below the limit of the current Technical Specifications. A lower required flow rate would speed up the surveillance procedures and help reduce downtime during refueling. A small break calculation using LVA with reduced HPSI flow shows a worst break Limit Value PCT of about 1450°F for a 20% reduction in HPSI flow. Assuming, again, an uncertainty of 150°F for the code-related uncertainties, the resulting Limit Value PCT with reduced HPSI flow of 1600°F is still lower than the Appendix K PCT of 1730°F with full HPSI flow.
FUTURE PLANS

Combustion Engineering, together with EPRI, has outlined a project to demonstrate the use of the Limit Value Approach of accounting for uncertainties due to plant conditions and parameters. It is planned to do this work with RELAP5/MOD2 as the LOCA computer program. The quantification of the code-related uncertainties will be based on the results from NRC/INEL work on this subject. In addition, it is planned to utilize the results from RELAP5/MOD2 assessment calculations in the framework of the International Code Assessment Program (ICAP). The overall goal of this effort is to demonstrate a realistic LOCA calculation and uncertainty estimation methodology which can be used with the proposed amendment to the LOCA/ECCS Rule and is economical as well. This methodology will be applied initially to small break LOCA calculations, later it will be extended to large break LOCA analyses. An additional portion of this project is a comparison of the Limit Value Approach with a statistical approach to account for uncertainties due to plant conditions and parameters. The intent of this effort is to identify the magnitude of the conservatism of the Limit Value Approach relative to a 95% probability level statistical approach.

ACKNOWLEDGEMENT

This paper describes the results of work which was performed by many people in the Transient Methods and LOCA Unit of Combustion Engineering. Their contributions are gratefully acknowledged.

References

1. CEN-373-P (Proprietary); Realistic Small Break LOCA Evaluation Model, Volume I, Calculational Models; April 1988.
2. NUREG-1230; Compendium of ECCS Research for Realistic LOCA Analysis; April 1987.
FIGURE 1

AMENDED LOCA/ECCS RULE - CALCULATION OF PCI AND UNCERTAINTIES

CSAU (NRC) APPROACH

LIMIT VALUE (C-F) APPROACH

REALISTIC PCI PLUS UNCERTAINTIES

ΔPCT-CODE

ΔPCT-SCALING

ΔPCT-DEFICIENCIES

ΔPCT-PLANT PARAMETERS

REALISTIC PCI WITHOUT UNCERTAINTIES
(REALISTIC CALCULATION MODEL, NOMINAL PLANT PARAMETER INPUT)

LIMIT VALUE PCI PLUS UNCERTAINTIES

ΔPCT-CODE

ΔPCT-SCALING

ΔPCT-DEFICIENCIES

LIMIT VALUE PCI WITHOUT UNCERTAINTIES
(REALISTIC CALCULATION MODEL, LIMIT VALUE PLANT PARAMETER INPUT)