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OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations

Workshop Proceedings
Barcelona, Spain
16-18 November 2011

Part 3

BETTER POLICIES FOR BETTER LIVES

NUCLEAR ENERGY AGENCY

Organisation de Coopération et de Développement Économiques
Organisation for Economic Co-operation and Development

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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

NEA/CSNI/R(2013)8/PART3
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OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 34 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Republic of Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation’s statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 31 countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, the Republic of Korea, the Russian Federation, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

The mission of the NEA is:

– to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as

– to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

THE COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

"The Committee on the Safety of Nuclear Installations (CSNI) shall be responsible for the activities of the Agency that support maintaining and advancing the scientific and technical knowledge base of the safety of nuclear installations, with the aim of implementing the NEA Strategic Plan for 2011-2016 and the Joint CSNI/CNRA Strategic Plan and Mandates for 2011-2016 in its field of competence.

The Committee shall constitute a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development and engineering, to its activities. It shall have regard to the exchange of information between member countries and safety R&D programmes of various sizes in order to keep all member countries involved in and abreast of developments in technical safety matters.

The Committee shall review the state of knowledge on important topics of nuclear safety science and techniques and of safety assessments, and ensure that operating experience is appropriately accounted for in its activities. It shall initiate and conduct programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach consensus on technical issues of common interest. It shall promote the co-ordination of work in different member countries that serve to maintain and enhance competence in nuclear safety matters, including the establishment of joint undertakings, and shall assist in the feedback of the results to participating organisations. The Committee shall ensure that valuable end-products of the technical reviews and analyses are produced and available to members in a timely manner.

The Committee shall focus primarily on the safety aspects of existing power reactors, other nuclear installations and the construction of new power reactors; it shall also consider the safety implications of scientific and technical developments of future reactor designs.

The Committee shall organise its own activities. Furthermore, it shall examine any other matters referred to it by the Steering Committee. It may sponsor specialist meetings and technical working groups to further its objectives. In implementing its programme the Committee shall establish co-operative mechanisms with the Committee on Nuclear Regulatory Activities in order to work with that Committee on matters of common interest, avoiding unnecessary duplications.

The Committee shall also co-operate with the Committee on Radiation Protection and Public Health, the Radioactive Waste Management Committee, the Committee for Technical and Economic Studies on Nuclear Energy Development and the Fuel Cycle and the Nuclear Science Committee on matters of common interest."

OVERVIEW OF OECD/NEA BEPU PROGRAMMES

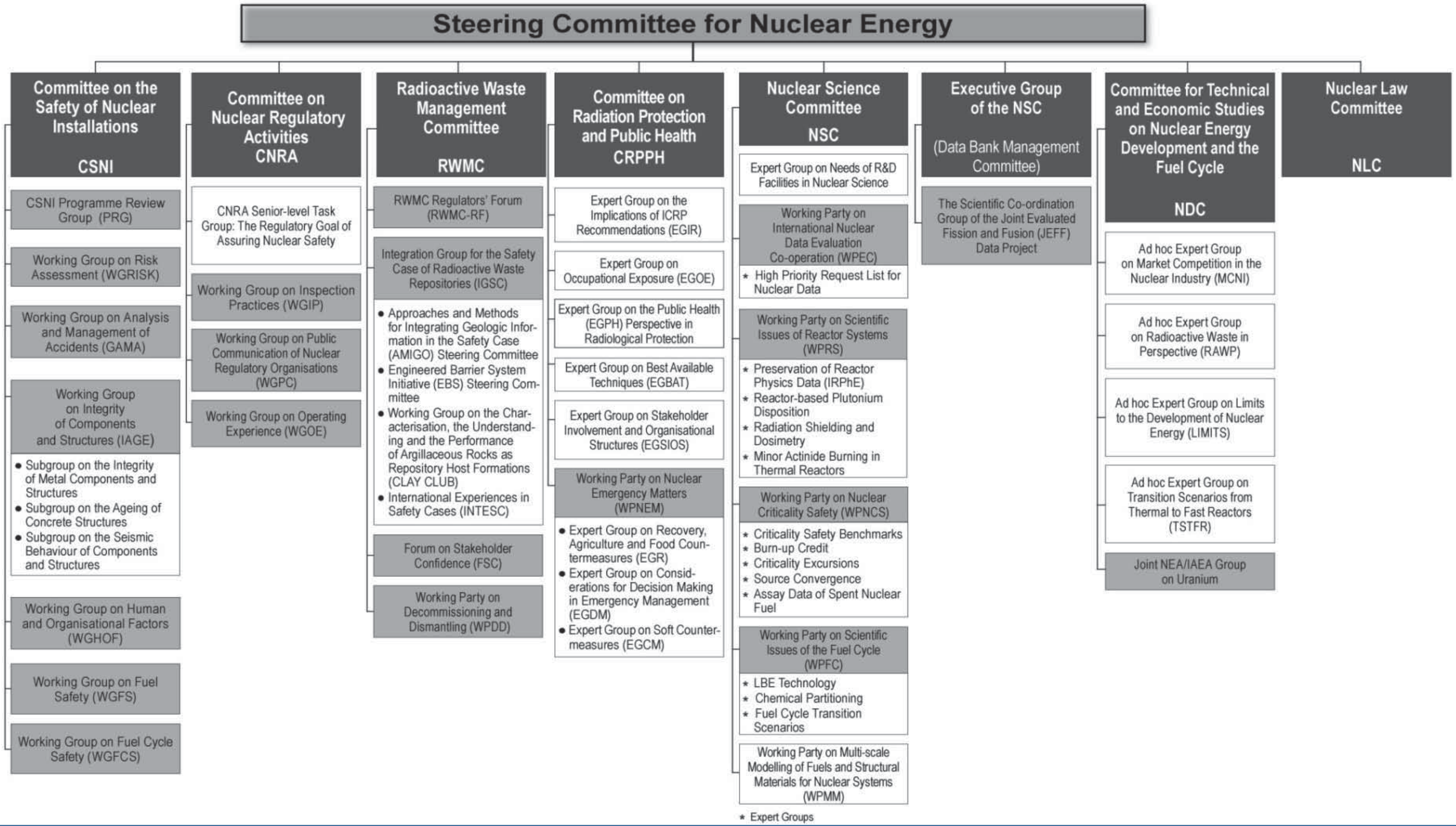
Abdallah AMRI, Jim GULLIFORD
OECD Nuclear Energy Agency

Outline

- NEA contribution to BEPU development and assessment
  - International Standard Problems (ISPs)
  - Benchmarks
  - Validation Matrices
  - OECD Joint Safety Research Projects
  - Specialist Meetings
- Examples of NEA BEPU related Programmes
  - Uncertainty Methods Study (UMS)
  - Best-Estimate Methods – Uncertainty and Sensitivity Evaluation (BEMUSE)
  - Safety Margin Assessment and Application (SM2A)
  - Uncertainty Analysis in Modeling (UAM) Benchmark
- Summary

NEA Structure

Committee structure of the OECD Nuclear Energy Agency (NEA)



BEPU PATH

IN THE 70s

- ECCS rule of 1974 recognizes limited state of knowledge and imposes/recommends conservatisms through Appendix K
  - Atomic Energy Commission directs research to be conducted to establish the magnitude of safety margins and alleviate conservatisms where indicated.
  - American Physical Society review of ECCS rule points out, among others, that without knowing where the "realistic" value is, one can never be sure that a prediction is conservative.

TODAY

- SM2A Pilot exercise
- SMAP Framework
- BEMUSE Conclusions
- IAEA SSG-2
- 10 CFR 50.46 and RG 1.157, RG 1.203

Today available options

Option	Applied Code	BIC	System availability	References
Conservative	Conservative code (Evaluation Model)	Conservative	Conservative assumptions	10 CFR § 50.46 (a)(1)(ii), Appendix K
Conservative	B-E code	Conservative	Conservative assumptions	IAEA Guide NS-G-1.2, 4.89; several other practices
B-E plus Uncertainty	B-E code + uncertainty evaluation	Realistic + uncertainty; partly most unfavourable conditions	Conservative assumptions	10 CFR § 50.46 (a)(1)(i), Appendix A. IAEA Guide NS-G-1.2, 4.90
Risk-informed	B-E code + uncertainty evaluation	Realistic + uncertainty	PSA-based assumptions	Draft change of US 10 CFR § 50.46. SMAP Framework.

Today available options

- A consistent BEPU application assumes:
  - Use of verified and validated computer code(s)
  - Use of a qualified uncertainty method.

- NEA, through its relevant Committees (CSNI and NSC) contributed by concrete tasks to the efforts of:
  - Code validation
  - Uncertainty method qualification
  - BEPU application according to the different options.

Code Validation

Code validation

- ISPs
- Benchmarks
- Validation Matrices
- OECD Joint Safety Projects
- Specialist Meetings

ISPs (1)

- ISPs triggered by the need to have an idea on how the Thermal-hydraulic codes were capable to simulate accidents

- A need to formalize the definition of an ISP appeared immediately
  - CSNI report N° 17
  - Revised 4 times, keeping the same goal
  - Last revision: NEA/CSNI/R(2004)5

- ISPs were first initiated in 1973 in the area of primary circuit Thermal-hydraulics, and then were progressively extended within CSNI to:
  - Containment TH
  - Fuel behaviour during a LOCA
  - Severe accidents.

ISPs (2)

- **50 ISPs so far, the last one recently completed and addressing a 50% DVI Line Break on ATLAS test facility**

- **ISPs contributions**
  - ISPs provide an important contribution to the code assessment process and are good candidates to be included in code validation matrices;
  - ISPs have been identifier of the user effect;
  - Benefits to the host organization (e.g., valuable comments and feedback from the international community; recognition and international consensus on the conclusions);
  - Benefits to the participants (e.g., privileged access to information on the experimental programme; a mean of performing code assessment, detailed discussion on several technical subjects);
  - Enhanced scientific discussion between code developers, users in different countries and experimentalists.

- **Forward looking**
  - Need to continue ISPs, in particular to address new designs (e.g., APR1400, AP1000).

Benchmarks (1)

Benchmark methodology

- Reference design from a real reactor
- Problem with a complete set of input data
- Three Benchmark phases
  - Phase 1: Point kinetics/ plant simulation
  - Phase 2: Coupled 3D Neutronics/ TH evaluation of core response
  - Phase 3: B-E coupled core/ plant transient model
- Evaluation of HZP and HFP steady states
- Simulation of best-estimate and extreme transient scenarios
- Method for comparison of results from different computer codes.

Benchmarks (2)

- **Three Benchmarks:**
  - OECD/NEA/NRC PWR MSLB Benchmark
  - OECD/NEA/NRC BWR TT Benchmark
  - OECD/DOE/CEA VVER-1000 CT Benchmark (based on actual Kozloduy 6 plant data):
    - V1000CT-1: main coolant pump start-up test
    - V1000CT-2: SG isolation experiment.

- **All the three Benchmarks completed and reports published**

- **The ongoing “Uncertainty Analysis in Modelling” (UAM) activity started as follow-up of these Benchmarks**
  - See below.

CSNI Validation Matrices (1)

- **Tasks initially given to the CSNI PWG-2**
  - To formulate an internationally agreed validation matrix by establishing cross reference matrices and selecting well balanced sets of experiments in the available database;
  - Data should follow the standard required for data use-ability set-up in the CSNI Report N°17.

- **Deliverables**
  - Start of the activity in 1983 with a report issued in March 1987 [CSNI Report N° 132]
  - SET Validation Matrix established between 1988 and 1993
  - Revision of the ITF Validation Matrix between 1993 and 1996 in “CSNI Integral Test Facility Validation Matrix for the Assessment of Thermal-hydraulic Codes for LWR LOCAL and Transients” [NEA/CSNI/R(96)17]
  - Validation Matrix for the Assessment of Thermal-hydraulic Codes for VVER LOCAL and Transients issued as [NEA/CSNI/R(2001)4]

CSNI Validation Matrices (2)

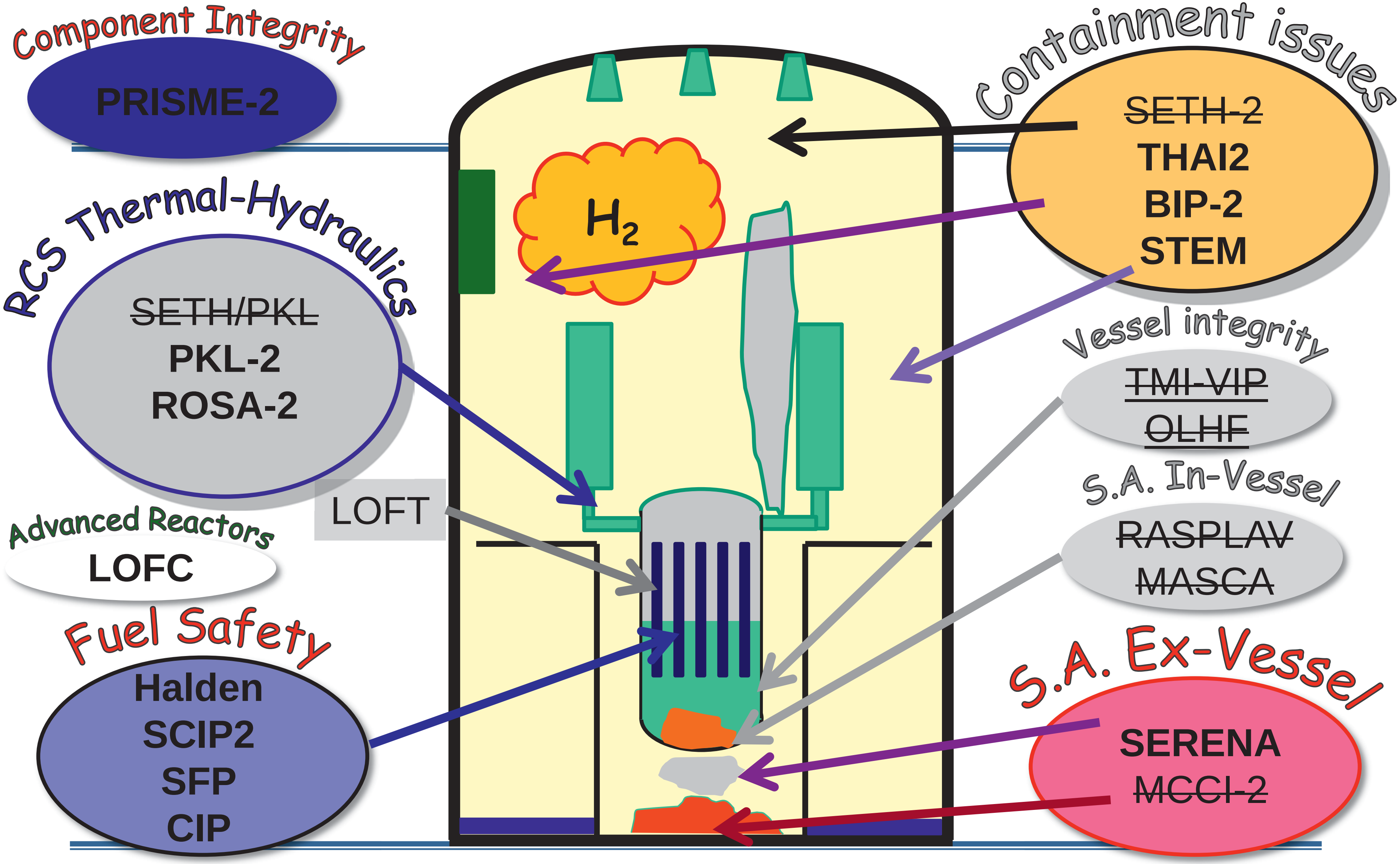
- **Outcome**
  - Internationally agreed validation matrices were established for TH system codes simulating PWR, BWR and VVER LOCA and transients.
    - Phenomena-based set of experiments defined
    - Include the major part of world wide experimental work in LWR TH safety research.
  - CSNI ITF and SET Validation Matrices used in establishing validation matrices for the major TH system codes;
  - The creation of the databases and the development of the TH system codes provided the components to implement BEPU methodologies.

JOINT SAFETY RESEARCH PROJECTS (1)

HALDEN	Fuel & Materials, I&C, HOF	Norway
CIP	Fuel in RIA transients in Cabri	France
SCIP-2	Fuel integrity	Sweden
SFP	Fuel hydraulics/ignition phenomena	USA
JHIP	Jules Horowitz international Program	France (proposed)
LOFC	Loss of Forced Coolant with HTTR	Japan (started)
PRISME-2	Fire safety	France
ROSA-2	System TH	Japan
PKL-2	PWR SG Heat Transfer	Germany
SETH-2	Containment TH (CFD)	Swit/Fra (compl.)
THAI-2	Containment (HTGR , H2, FP)	Germany (prop.)
BIP	Iodine chemistry	Canada
STEM	Source Term Evaluation & mitigation	France (proposed)
MCCI-2	Severe Accident (Ex-Vessel)	USA (completed)
SERENA	Steam explosion	Korea & France

- **Databases** :
- **1.FIRE 2.ICDE 3.OPDE/CODAP 4.COMPSIS 5.CADAK**

OVERVIEW OF OECD/NEA BEPU PROGRAMMES



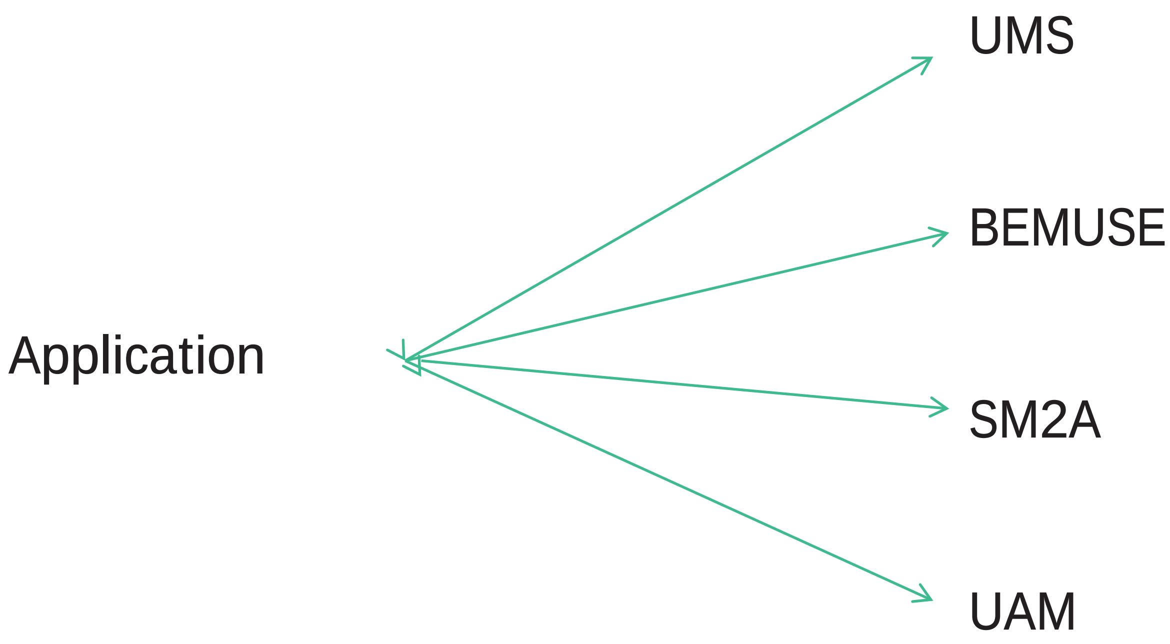
COMPLETED PROJECTS - SYNTHESIS SUMMARY REPORT UNDER PREPARATION

THAI	Containment (H2, FP)	Germany	2007-2009
ROSA	RCS Thermal-hydraulics	Japan	2005-2009
SCIP	Cladding Integrity	Sweden	2004-2009
PKL	Boron dilution	Germany	2004-2007
Paks-Phase1	Fuel damage	Hungary	2004-2007
PSB-VVER	VVER Thermal –Hydraulics	Russia	2003-2008
MCCI	Corium-concrete interaction	USA	2002-2006
SETH-Panda	Containment TH	Switzerland	2001-2006
SETH-PKL	RCD Thermal-hydraulics	Germany	2001-2004
MASCA	Severe Accident (In Vessel)	Russia	2000-2006
O-LHF	Lower Head Failure	USA	1998-2002
BubCon	VVER TH	Hungary	1998-2002
PLASMA	VVER I&C	Hungary	1998-2002
SCORPIO	VVER I&C	Norway	1996-1998
RASPLAV	Severe Accident	Russia	1994-2000
TMI2-VIP	Pressure Vessel Inspection	USA	1988-1993
LOFT	LOCA + FP release	USA	1983-1989

JOINT SAFETY RESEARCH PROJECTS (4)

- **Provide**
  - A sound framework for computer code assessment
  - Well defined experiments and well documented test results for code validation
  - Useful exchanges between experimentalists and code developers/users
  - A useful framework for knowledge transfer, especially to new generation of code users.

BEPU Application



Emergence of the uncertainty concept (1)

- **Some steps**
  - Analysis of ISP 18 (LOBI) – June 1986
  - Ad hoc meeting on code uncertainties in Wurenlingen in September 1986
  - Presentation of CSAU by N. Zuber – June 1987
  - Exercise on uncertainty evaluation performed by UK and GRS leading to formalization of the British Method (BM) and the GRS method (GM)
  - Discussion of a comparison of BM, GM and the CSAU methods – February 1988
  - Presentation by UNIPI of FFT meteorologist methods to measure the code accuracy – June 1988
  - Presentation by France of the ASM (Adjoint Sensitivity Method) – February 1991
  - Applications of the different methods to PHEBUS, OMEGA, SBLOCA, ISP 27 – between 1990 and 1993.

Emergence of the uncertainty concept (2)

- Following similar meetings in Toronto (1976), Paris (1978), Pasadena (1981), organization of the Transient Two Phase Flow Meeting in Aix-en-Provence (April 1992) to discuss the status of advanced codes, in particular their application and assessment of uncertainties in code calculations
- Presentation by UNIPI of the UMAE (Uncertainty Methodology based on Accuracy Extrapolation) – July 1993
- CSNI Workshop on Uncertainty Analysis Methods (London, 1-4 March 1994)

- Several discussions on how to organize an ISP on uncertainties
- Consensus obtained on a proposal of ISP called Uncertainty Methods Study (UMS) exercise to compare the uncertainty methods, step by step, on the same problem and comparison with measured values of the LSTF SBCL 18.

The UMS Exercise (1)

- Subject: Analysis of LSTF SB-CL-18 (5% cold leg break)- Prototype investigation in evaluating uncertainties and comparing the contributions step by step.
- Objectives approved by CSNI in December 1994:
  - To gain insights into differences between method features
  - To inform decision makers on conducting uncertainty analyses, e.g., in the light of licensing requirements.
- Period: May 1995 – June 1997
- Report on the Uncertainty Methods Study [NEA/CSNI/R(97)35]
- Participating organizations:
  - AEA Technology, UK: RELAP5/MOD3.2, AEAT uncertainty method
  - UNIPI, Italy: RELAP5/MOD2, CATHARE 2 V1.3U rev5, UMAE
  - GRS, Germany: ATHLET Mod 1.1 Cycle A, GRS method
  - IPSN, France: CATHARE 2 V1.3U rev5, IPSN method
  - ENUSA, Spain: RELAP5/MOD3.2, ENUSA method.

The UMS Exercise (2)

- Main conclusion: The way the different methods are applied was very important
- Choice of the methods:
  - Each UMS participant favored the applied method
  - In all cases, appropriate knowledge, skill, experience and quality standards had to be applied
- The differences between the predictions of the methods came from a combination of:
  - The method used and the way to use it;
  - The accuracy of the reference calculation and the modeling used
  - The completeness of the identification and selection of uncertainties
  - The conservatism of the calculation input (e.g., uncertainty ranges or probability distributions)
  - Optimization of the nodalization.
- See detailed presentation in Paper S2.1

The BEMUSE Programme (1)

Background

- Proposals on B-E methods and applications
- Discussion during WGAMA meetings 1,2 and 3,
- Then during the Exploratory Meeting of Experts to define an Action Plan on B-E Calculations and Uncertainty Analysis (Aix-en-Provence, 13-14 May 2002)
- Agreement in September 2002 on the Action Plan

BEMUSE (Best-Estimate Methods – Uncertainty and Sensitivity Evaluation) Programme objectives:

- To evaluate the practicability, quality and reliability of B-E methods including uncertainty evaluation in applications relevant to Nuclear Reactor Safety (NRS)
- To develop common understanding
- To promote and facilitate the use of BEPU methods by the safety organizations and by the industry.

The BEMUSE Programme (2)

Step 1

Phase I: Presentation of the uncertainty methods

Phase II: Re-analysis of LOFT L2-5 experiment

Phase III: Uncertainty evaluation of the L2-5 test calculation - Conclusions on method **BEMUSE** improvement

Step 2

Phase IV: B-E analysis of the LBLOCA on plant scale (Zion)

Phase V: Sensitivity studies and uncertainty evaluation of the NPP LBLOCA

Phase VI: Status report, classification of the methods –Concl. and recom.

Step 2

The BEMUSE Programme (3)

- Schedule
  - Step 1: January 2004 – May 2006
  - Step 2: August 2006 – September 2010
- Participants
  - 14 participants from 10 countries
  - Not all participants were involved in all phases
- Computer codes used
  - ATHLET, CATHARE, MARS, RELAP5, TECH-M-97, TRACE
- Uncertainty methods used – 2 types:
  - Statistical method with propagation of input uncertainties to output uncertainties by code calculations
  - UMAE/ CIAU (Uncertainty Method based on Accuracy Extrapolation/ Code with Capability of Internal Assessment of Uncertainty)

The BEMUSE Programme (4)

Overall conclusions from Phase VI

- BEPU used may be considered mature for application, including in licensing process;
- Differences observed in application of the methods which lead to different results, even among the base calculation results;
- Importance of user effect in the base case and in the application of uncertainty methods;
- Effort should be focused on the base case, on the influential parameters, and on the distribution of the uncertain input parameters and their range;
- Method(s) to select and quantify computer code model uncertainties and to compare their effect on the uncertainty on the results to be performed in the frame of an international benchmark using different computer codes.

See detailed presentation in Paper S2.2

PREMIUM Benchmark – BEMUSE Follow-up

- **PREMIUM Benchmark** : activity just starting in order to address recommendations of BEMUSE Phase VI.
  - **Objective**: Use the measured data of an analytical reflood experiment in order to derive the uncertainties of physical models (e.g., heat transfer downstream from the quench front, relative velocities upstream or downstream from the quench front)
  - **Programme:** Five phases which will be completed in spring 2014:
    - Phase I : Description of the existing methods
    - Phase II: Identification of influential input parameters
    - Phase III: Determination of the ranges of variation of the multipliers p of the considered physical models PM (PM = pxPMnominal) on the basis of qualified experimental results
    - Phase IV: Confirmation of the ranges of variation found in Phase III by using PERICLES-2D experimental results
    - Phase V: Final synthesis report, including conclusions and recommendations.
  - See detailed presentation in Paper S2.4

Safety Margin Assessment and Application (SM2A) SMAP Framework (1)

- **SMAP Framework Objectives**
  - To agree on a framework for integrated assessments of the changes to overall safety of the plant as a result of simultaneous changes in plant operation/condition
  - To develop a CSNI document which can be used by Member countries to assess the effect of plant change on the overall safety of the plant
  - To share information and experience
- **SMAP Framework overview**
  - Action Plan distributed in 5 tasks
  - 20 experts from 15 countries participated
  - 7 Meetings held from October 2003 to October 2006
  - 4 Technical Notes issued in 2005 and 2006
  - Final Report [NEA/CSNI/R(2007)9] issued in 2007

Safety Margin Assessment and Application (SM2A) SMAP framework (2)

How to Quantify Global Plant Safety Margin?

- Use existing tools and techniques
- Merge
  - Deterministic approach: accepted definition of safety margins in the nuclear industry
  - Probabilistic risk assessment: include all relevant accident sequences
- Develop risk metrics (e.g., ΔCDF and ΔLERF) that can be used to evaluate a plant modification against existing regulatory acceptance criteria and guidelines

Safety Margin Assessment and Application (SM2A) SMAP Framework (3)

➔ **Steps (perform all before and after the plant modification):**

Likelihood that event sequence will occur & Conditional probability that the core will loose function

from event tree frequency

from engineering data, safety limits and deterministic calculations (the CPLF)

1. Decide on uncertainties in the deterministic calculations for the particular safety margin
2. Complete best estimate plus uncertainty calculation
3. Multiply frequency with exceedance probability
4. Add over all event sequences to get cumulative core damage frequency

Safety Margin Assessment and Application (SM2A)
SM2A Pilot Exercise (1)

- Task Group decided during CSNI meeting June 2007:
  - To appraise SMAP methodology using US proposed new LOCA rulemaking as test case
  - Preliminary results to be reported to CSNI in June 2009 → short/focused activity
- Mandate prepared by NEA and sent to CSNI August 1st, 2007
- Discussion during CSNI meeting December 2007: Tight and ambitious schedule raised as a concern
  - Nominations received from 9 countries (+IAEA)
  - First meeting held at US NRC Offices, January 17-18, 2008

Safety Margin Assessment and Application (SM2A)
SM2A Pilot Exercise (2)

SMAP framework implementation

- Short-lived and focused appraisal of SMAP methodology required, <u>not real-life application</u>
- Application should reflect multiple changes, including plant changes

Hypothetical 10 % Power Up-rate for Zion PWR

- Decommissioned plant without sister plants
- Was studied in NUREG-1150
- Some PSA documentation available (event trees)
- Many participants already have input deck from BEMUSE exercise.

Safety Margin Assessment and Application (SM2A)
SM2A Pilot Exercise (3)

Type	Org	# Seq.	Code	Org
LBLOCA	EDF	2	CATHARE/TRACE	EDF/NRI
MBLOCA	PSI	1	TRACE	PSI
SBLOCA	PSI	1	TRACE	PSI
LOSP	IRSN/CNSNS	5	CATHARE	IRSN
MSLB	STUK	2	ATHLET	GRS
SGTR	KAERI/KINS	3	MARS	KAERI
TT	JNES	5	RELAP5	JNES
LOFW	NRC	1	TRACE	US NRC
L CC/SW	CSN	Damage domain	MAAP (TRACE)	CSN
Total	**SM2A**	**20 +**		

Safety Margin Assessment and Application (SM2A)
SM2A Pilot Exercise (4)

- Overall conclusions
- SMAP framework was proven workable for evaluation of safety margins.
  - Some refinements (screening of PSA sequences, reformulation of existing event trees) were however needed
- Increase of probability of exceedance for surrogate limit (PCT) indicating core damage was successfully evaluated for chosen scenarios from several Event Trees.
  - Impact of power up-rate could also be traced for scenarios with no criteria violation.
- Other conclusions drawn-up in terms of lessons learned, limitations of the SM2A exercise and possible improvements
- See detailed presentation and conclusions in Paper S4.2

OECD LWR Uncertainty Analysis in Modeling (UAM) Benchmark (1)

- Main motivations
- Principles supporting Risk-informed regulation should be considered in an integrated decision-making process
- Hence, any evaluation of licensing issues supported by a safety analysis should take into account both deterministic and probabilistic aspects of the problem
- The deterministic aspects should be addressed using BEPU
- Increasing demand from nuclear research, industry and safety organizations for B-E predictions to be provided with their confidence levels
- In the OECD LWR UAM Benchmark, uncertainty propagation is being evaluated through the whole simulation process in a unified benchmark framework to provide coupled code predictions with uncertainty evaluations of safety margins at the full core/ system level.

OECD LWR Uncertainty Analysis in Modeling (UAM) Benchmark (2)

Overall description

- Benchmark framework based on 9 steps (or exercise) grouped in 3 Phases.
- For each exercise, Input (I), Output (O), and target Uncertainty (U) parameters are identified
- When identifying the source of Input (I) uncertainties for each Exercise, which input uncertainties are propagated from the previous Exercise and which ones are new?
- Other important parameters to be defined are Output (O) uncertainties and propagated Uncertainty parameters (U) for each exercise.
  - The Output (O) uncertainties are used, for specified output parameters for each Exercise, to evaluate the used uncertainty method.
  - The propagated Uncertainty parameters (U) are output parameters which selected to be propagated further through the follow-up Exercises.

OECD LWR Uncertainty Analysis In Modeling (UAM) Benchmark (3)

UAM

- Phase I (Neutronics Phase)
  - Exercise I-1: Cell Physics
  - Exercise I-2: Lattice Physics
  - Exercise I-3: Core Physics
- Phase II (Core Phase)
  - Exercise II-1: Fuel Physics
  - Exercise II-2: Neutron kinetics
  - Exercise II-3: Bundle TH
- Phase III (System Phase)
  - Exercise III-1: Core Multi-Physics
  - Exercise III-2: System TH
  - Exercise III-3: Coupled Core-System

OECD LWR Uncertainty Analysis In Modeling (UAM) Benchmark (4)

Participation

- Participants can participate in the 3 Phases and in all exercises; alternatively they can participate in selected exercises
- There are 3 types of operating LWRs to be followed in this Benchmark: BWR (PB-2), PWR (TMI-1) and VVER (Kozloduy-6 and Kalinin-3)
- Participants can model one or more reactor types depending on their interest
- For each Exercise, two types of test problems are designed: numerical test problem provided with reference solutions and experimental test problems obtained from publicly available databases.
- See detailed presentation in Paper S2.3, including status and results of Phase I, status of Phase II and priorities of Phase III.

Summary

- **The OECD/NEA paved the way for the development and assessment of BEPU for about 40 years, through concrete tasks:**
  - ISPs, Benchmarking activities
  - Development of Validation Matrices
  - Joint Safety Research Projects
  - Specialist meetings
- **Several NEA related BEPU programmes have been successfully completed:**
  - Uncertainty Methods Study (UMS)
  - Best-Estimate Methods – Uncertainty and Sensitivity Evaluation (BEMUSE)
  - Safety Margin Assessment and Application (SM2A)
- **New Programmes are underway to address pending issues (e.g., input uncertainties, uncertainties in coupled codes)**
- **The present Workshop may highlight new issues to be addressed (e.g., uncertainty analysis for CFD codes).**

Thank you for your attention.

Any question?

International Atomic Energy Agency

Keynote Paper

Best Estimate plus Uncertainty (BEPU) Analyses in the IAEA Safety Standards

Milorad Dusic
IAEA, Division of Nuclear Installation Safety

OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations

Barcelona, Spain

16 – 18 November 2011

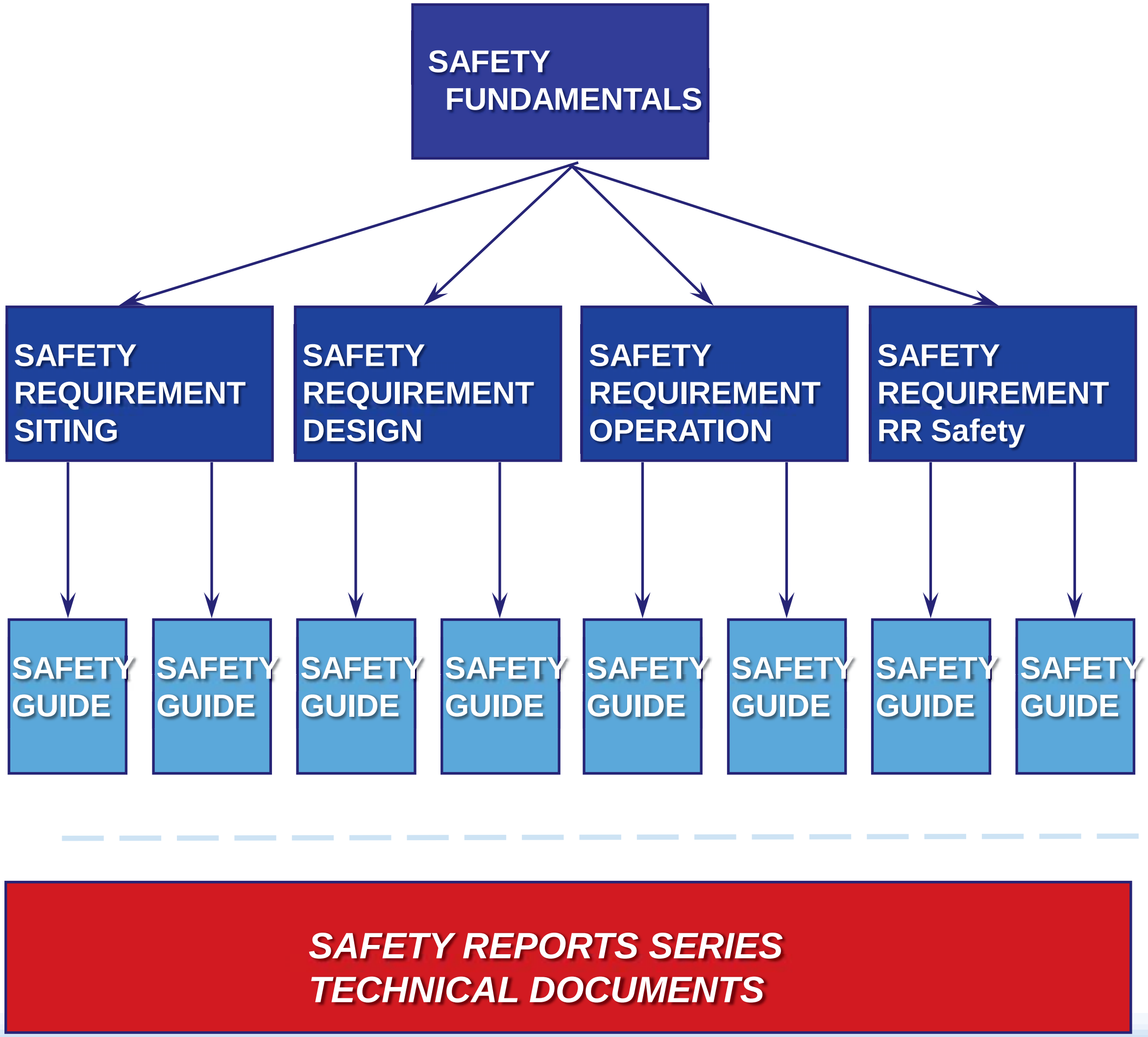
SAFETY STANDARDS SERIES

- **Safety Standards Series publications are categorized into:**
  - **Safety Fundamental (F; blue lettering)**
  - **Safety Requirements (R; red lettering)**
  - **Safety Guides (G; green lettering)**

SF-1

In late 2006 the IAEA published:

HIERARCHY OF THE IAEA SAFETY STANDARDS SERIES



Present the overall objectives, concepts and principles of protection and safety. They are the policy documents of the safety standards

Establish requirements that must be met to ensure the protection and safety of peopl and the environment, both now and in the fu

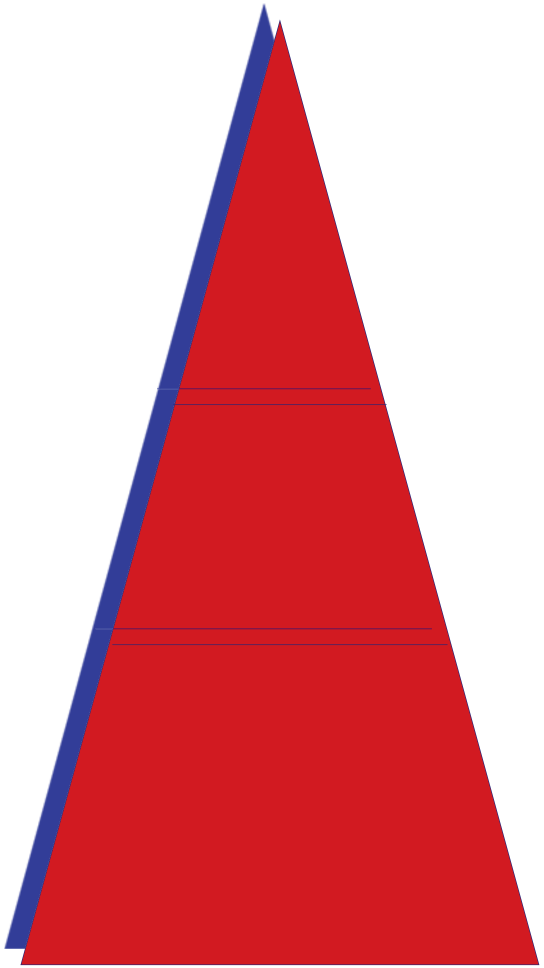
Provide guidance, in the form of more detail actions, conditions or procedures that can b used to comply with the Requirements

Practical examples and detailed methods for the application of the Safety Standards. Detailed Technical Reports

International Atomic Energy Agency

Fundamentals

The Two Conventions



Requirements

National Safety Regulations

Guides

National Regulatory Guides

New Safety Standards Structure

Safety Fundamentals
Fundamental Safety Principles

General Safety Requirements

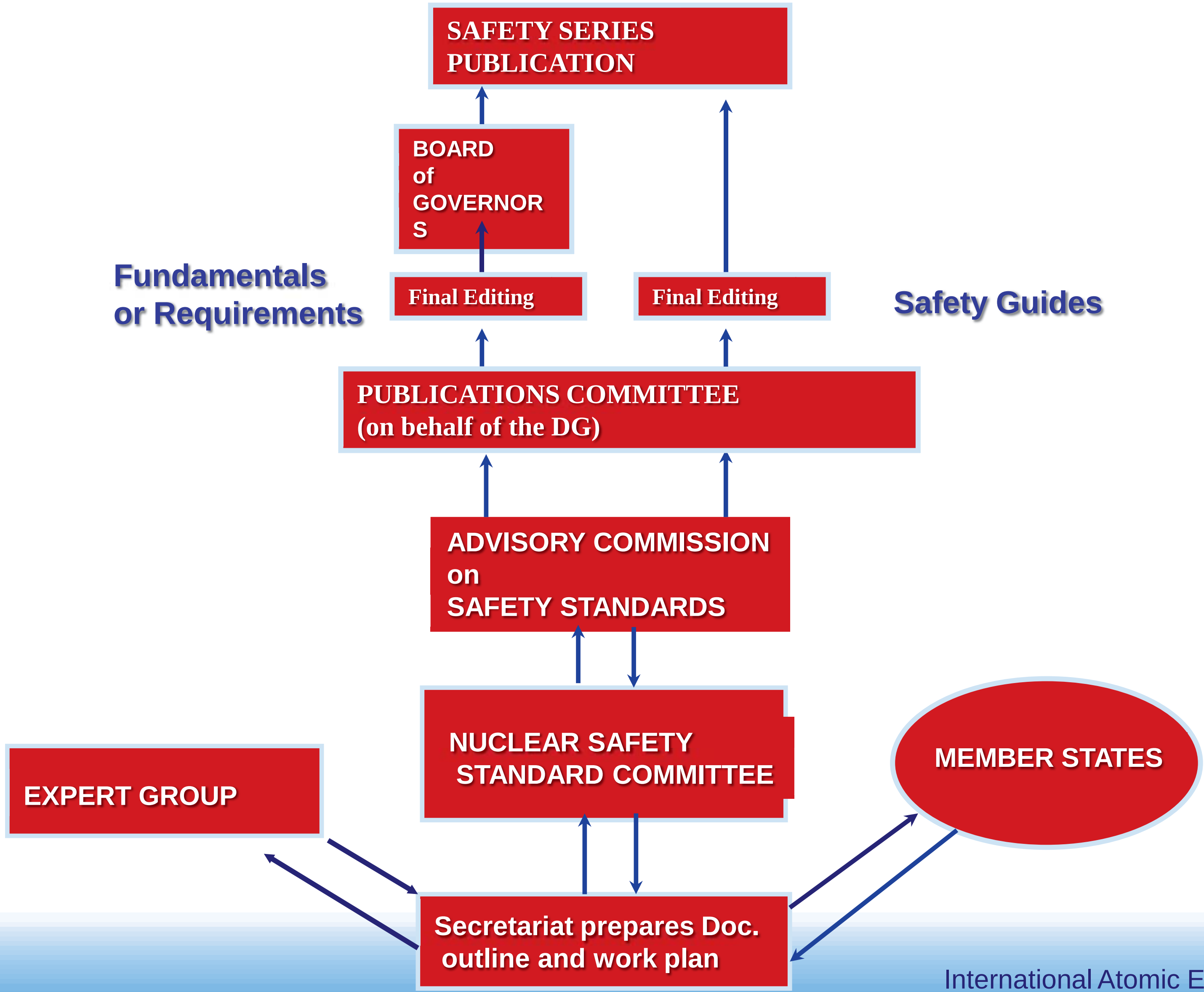
- Part 1. Governmental, Legal and Regulatory Framework for Safety
- Part 2. Leadership and Management for Safety
- Part 3. Radiation Protection and the Safety of Radiation Sources
- Part 4. Safety Assessment for Facilities and Activities
- Part 5. Predisposal Management of Radioactive Waste
- Part 6. Decommissioning and Termination of Activities
- Part 7. Emergency Preparedness and Response

Specific Safety Requirements

1. Site Evaluation for Nuclear Installations
2. Safety of Nuclear Power Plants
   - 2.1. Design and Construction
   - 2.2. Commissioning and Operation
3. Safety of Research Reactors
4. Safety of Nuclear Fuel Cycle Facilities
5. Safety of Radioactive Waste Disposal Facilities
6. Safe Transport of Radioactive Material

Collection of Safety Guides

SAFETY STANDARDS REVIEW PROCESS



International Atomic Energy Agency

I. BEPU in the IAEA Safety Standards

- **Safety Requirements SSR 2/1: Safety of NPPs; Design (Revision of NS-R-1)**
- **General Safety Requirement GSR Part 4: Safety Assessment for Facilities and Activities**
- **Safety Guide SSG-2 Deterministic Safety Analysis for Nuclear Power Plants**

SSR – 2/1 Safety of NPPs; Design

- **Requirement 19: Design Basis Accidents**

Design Basis Accidents shall be analysed in a conservative manner

- **Requirement 20: Design Extension Conditions**

An analysis of design extension conditions shall be performed with best estimate approach (more stringent approaches may be used according to States' requirements)

- **Requirement 42: Safety Analys. of Plant Design**

A saf. analys. of the design shall be conducted in which methods of both det. and prob. analyses shall be applied

GSR Part 4: Safety Assessment of Facilities and Activities

- **Requirement 15: Det. and prob. Approaches**

Both deterministic and probabilistic approaches shall be included in the safety analysis

- **Requirement 16: Criteria for judging safety**

Criteria for judging safety shall be defined for the safety analysis

GSR Part 4 – cont.

- **Requirement 17: Uncertainty and sensitivity analysis**

Uncertainty and sensitivity analysis shall be performed and taken into account in the results of safety analysis and the conclusions drawn from it

Definitions of aleatory (or stochastic) and epistemic uncertainties are given

- **Requirement 18: Use of computer codes**

Any calculationel methods and computer codes used in safety analysis shall undergo verification and validation

4 OPTIONS in SSG-2

Applied codes	Input & BIC (boundary and initial conditions)	Assumptions on systems availability	Approach	Regulation
Conservative codes	Conservative input	Conservative assumptions	Deterministic*	10 CFR 50.46 Appendix K
Best Estimate (realistic) codes	Conservative input	Conservative assumptions	Deterministic	SG NS-G-1.2 para 4.89
Best estimate (realistic) codes	Realistic input + Uncertainty	Conservative assumptions	Deterministic	SG NS-G-1.2 para 4.90
Best estimate (realistic) codes	Realistic input + Uncertainty	PSA-based assumptions	Deterministic + probabilistic	Risk informed

NUSSC suggestion from their last meeting:

NUSSC suggested that new safety guides should be accompanied by documents like TECDOCs or Safety Reports describing in detail their recommendations where appropriate.

SRS #52

SRS # 52 - Best Estimate Safety Analysis for NPPs: Uncertainty Evaluations

- Overview of Uncertainty Methods
- Qualification of Evaluation Methods
- Suggestions for Application of Methods
- Current Trends
- Conclusions

- Main Authors: D'Auria, Glaeser, Misak, Schultz

SRS # 52

Overview of Uncertainty Methods

- Probabilistic methods
  - CSAU
  - GRS
  - IPSN
  - ENUSA
  - GSUAM
  - BEAU

SRS # 52

Overview of Uncertainty Methods

- Deterministic methods
  - AEAW
  - Method used by EDF-Framatome

SRS # 52

- **ANNEX I: Sources of Uncertainties**
  - **Code or model uncertainties**
  - **Representation uncertainties**
  - **Scaling uncertainties**
  - **Plant uncertainties**
  - **User effect**
- **ANNEX II: Description of Methods and Examples of Results**
- **ANNEX III: Supporting Methods**
- **ANNEX IV: Examples of Licensing Applications**

II. Safety Report Series (SRS)

- SRS No. 23 Accident Analysis for NPPs
- SRS No. 29 Accident Analysis for NPPs with Pressurized Heavy Water Reactors
- SRS No. 30 Accident Analysis for NPPs with Pressurized Water Reactors
- SRS No. 32 Implementation of Accident Management Programs in NPPs
- SRS No. 43 Accident Analysis for NPPs with Graphite Moderated Boiling Water RBMK Reactors
- SRS No. 48 Development and Review of Plant Specific Emergency Operating Procedures
- SRS No. 52 Best Estimate Safety Analysis for NPPs: Uncertainty Evaluation

III. TECDOCs

- IAEA TECDOC - 1332 Safety Margins of Operating Reactors; Analysis of Uncertainties and Implications for Decision Making
- IAEA TECDOC - 1351 Incorporation of Advanced Accident Analysis Methodology into Safety Analysis Reports
- IAEA TECDOC - 1352 Application of Simulation Techniques for Accident Management Training in NPPs
- IAEA TECDOC - 1379 Use of Computational Fluid Dynamics Codes for Safety Analysis of Nuclear Reactor Systems
- IAEA TECDOC - 1418 Implications of Power Uprates on Safety Margins of NPPs
- IAEA TECDOC - 1440 Overview of Training Meth. for Accident Management at NPPs
- IAEA TECDOC - 1539 Use and Development of Coupled Computer Codes for the Analysis of Accidents at NPPs
- IAEA TECDOC - 1550 Deterministic Analysis of Operational Events in NPPs
- IAEA TECDOC - 1578 Computational Analysis of the Behaviour of Nuclear Fuel Under Steady State, Transient and Accident Conditions
- IAEA TECDOC - 1594 Analysis of Severe Accidents in Pressurized Heavy Water Reactors

Impact of the Fukushima Accident on SS
IAEA Ministerial Conference on Nuclear Safety
Vienna, Austria
20-24 June 2011

• Total number of registered participants:	1052
• No. of Member States registered:	124
• No. of UN and specialized Agencies	9
• No. of NGOs:	3
• No. of Ministers:	29
• No. of Journalists:	200
• No. of Statements MSs/International Organizations	83/14

Ministerial Declaration

25 Points

- Sympathy and solidarity with Japan
- **IAEA Safety Standards**
- Responsibility of Member States
- Central Role of IAEA in promoting international cooperation
- Need for comprehensive assessment of Fukushima accident
- Importance of IAEA International Peer Reviews
- Need for comprehensive risk and safety assessment of all NPPs

Safety Standards specific statements:

- Emphasize the importance of implementing enhanced national and international measures to ensure that **the highest and most robust levels of nuclear safety are in place, based on IAEA safety standards**, which should be continuously reviewed, strengthened and implemented as broadly and effectively as possible and commit to increase bilateral, regional and international cooperation to that effect;

Request to the CSS

- **CSS was asked to review** the relevant standards and to report within 12 months, with recommendations for strengthening them

Action Plan for Safety Standards that might need review in the future following the Fukushima Daiichi accident

- Topical Areas
  - Site Evaluation
  - Design of Nuclear Power Plants
  - Storage of Spent Fuel
  - Operational Safety, including Periodic Safety Review
  - Severe Accident Management
  - Emergency Preparedness and Response
  - Radiation Protection
  - Remediation
  - Transport Safety
  - Regulatory Control

PRIORITIZATION for the review: Review of the Safety Requirements first:

- NS-R-3 Site Evaluation for Nuclear Installations (2003)
- Draft DS 414 Safety of Nuclear Power Plants: Design
- Draft DS 413 Safety of Nuclear Power Plants : Commissioning and Operation
- GS-R-2 Preparedness and Response for a Nuclear or Radiological Emergency (2002)
- Draft DS 379 on Radiation Protection and Safety of Radiation Sources
- TS-R-1 Regulations for the Safe Transport of Radioactive Material (2009 Edition)
- GSR Part 1 Governmental, Legal and Regulatory Framework for Safety (2010)

Second Step: Review of Selected Safety Guides

- Draft DS 433 on Site Survey and Site Selection for Nuclear Installations prepared to revise 50-SG-S9 Site Survey for Nuclear Power Plants (1984)
- SSG-9 Seismic Hazards in Site Evaluation for Nuclear Installations (2010)
- NS-G-3.5 Flood Hazard for Nuclear Power Plants on Coastal and River Sites (2003)
- NS-G-2.13 Evaluation of Seismic Safety for Existing Nuclear Installations (2009)
- NS-G-3.6 Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plant (2004)
- Draft DS 430 prepared to revise NS-G-1.8 Design of Emergency Power Systems for Nuclear Power Plants (2004)
- DS 431 Design of I & C Systems for NPPs
- DS 371 Storage of Spent Fuel recently approved for publication
- DS 441 Construction of Nuclear Installations
- DS 413 Safety of Nuclear Power Plants: Commissioning and Operation
- DS 426 to revise NS-G-2.10 Periodic Safety Review of Nuclear Power Plants (2003)
- NS-G-2.15 Severe Accident Management Programmes for Nuclear Power Plants (2009)

CONCLUSIONS

- The Safety Standards Series establishes an essential basis for safety and represents the broadest international consensus.
- The incorporation of more detailed requirements, in accordance with national practice, may still be necessary.
- There should be only one set of international safety standards.
- Each safety standard will be reviewed by the relevant committee or by the commission every five years.
- Special review is currently underway to identify needs for revision in the light of the Fukushima accident.

Summary of Existing Uncertainty Methods

Paper S1.1

Horst Glaeser

OECD/ CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations
Technical University of Catalonia, Barcelona, Spain

16 – 18 November 2011

Contents

- **Main methods**

  1. Propagation of input uncertainties

  a) CSAU method (USA)

  b) Statistical methods

  - GRS Method (Germany), AREVA Method (USA), ASTRUM (Westinghouse, USA), GE (USA), KREM (Korea), KINS-REM (Korea), ESM-3D (France)
  - Number of code calculations – Wilks' formula
  - Number of calculations to meet more than one regulatory limit

  2. Extrapolation of output uncertainties

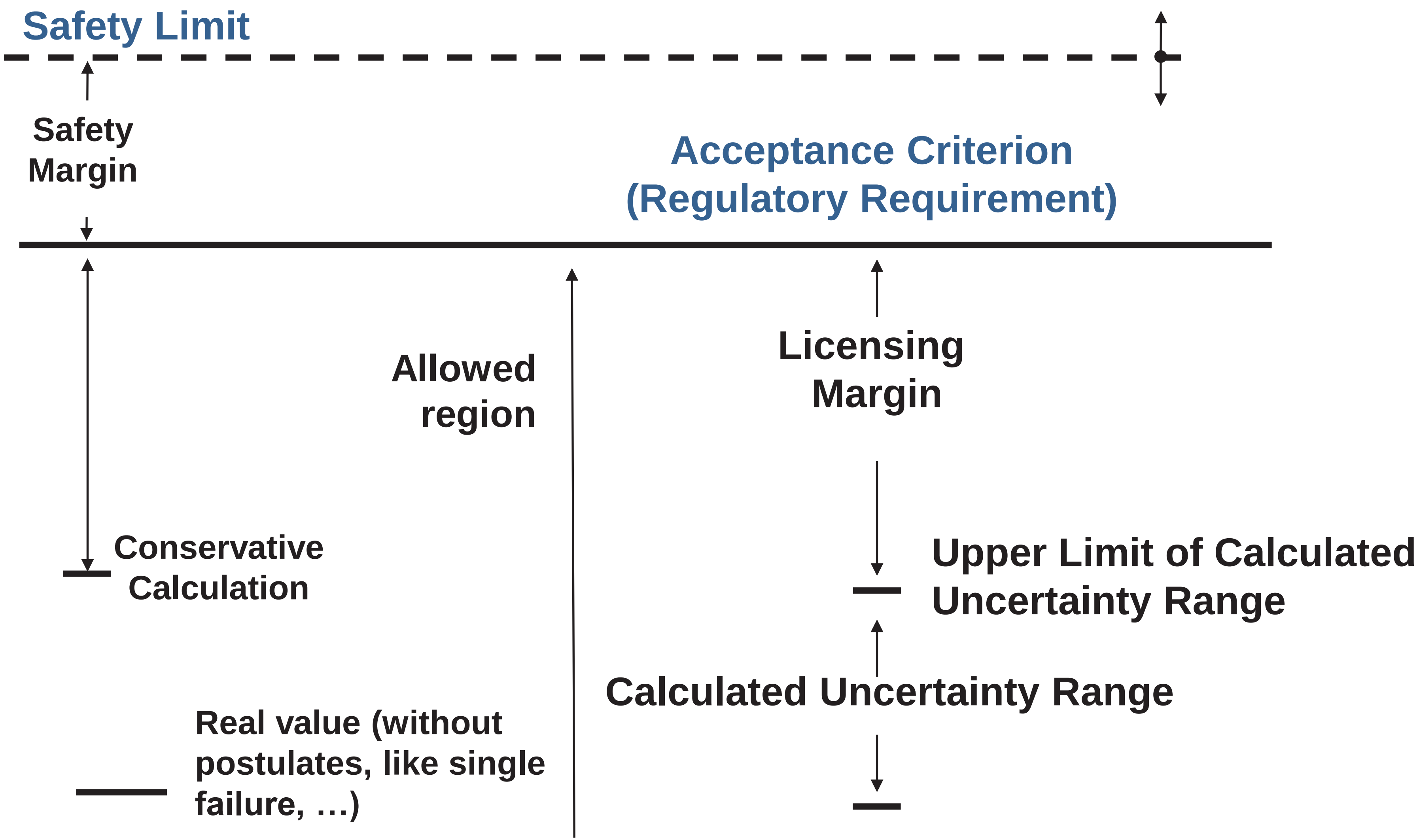
  - UMAE/ CIAU method (University Pisa)

- **Comparison of main methods**
- **Applications**
- **Conclusions**

Safety analysis of nuclear reactor steam supply systems

- To demonstrate that the plants are designed to respond safely to various postulated design basis accidents
- Performed by **computer simulation** using complex system codes due to significant variations of conditions that will occur during such an accident
- Models of thermal-hydraulic computer codes **approximate** the physical behaviour, and the solution methods are approximate due to **compromise of accuracy and calculation time**
  => Code calculation results are not exact but uncertain
- Uncertainties are taken into account by
  - **conservative** evaluation model calculations
  - **"best estimate" code** plus **conservative initial and boundary conditions**
  - **"best estimate" calculations** supplemented by **uncertainty analysis** of code results => **Uncertainty analysis method needed**

Illustration of Margins



CSAU (Code Scaling Applicability Uncertainty) Method (1)

- One of the first uncertainty methods proposed in the year 1988
- CSAU provides a **framework** to proceed through different steps in the process of evaluating uncertainty
- Investigate uncertainty of safety related single valued parameters, e.g. peak cladding temperature (PCT) or vessel water inventory
- Evaluation of the **code applicability** to a selected plant scenario
- Experts identify and rank phenomena by means of a process identification and ranking table (**PIRT**) to **select highly important phenomena**
- Single parameter sensitivity calculations performed using an **optimised nodalisation** capturing important physical phenomena
- Information from experiments, manufacturing, and validation calculations utilised for defining ranges and probability distributions of the uncertain input parameters

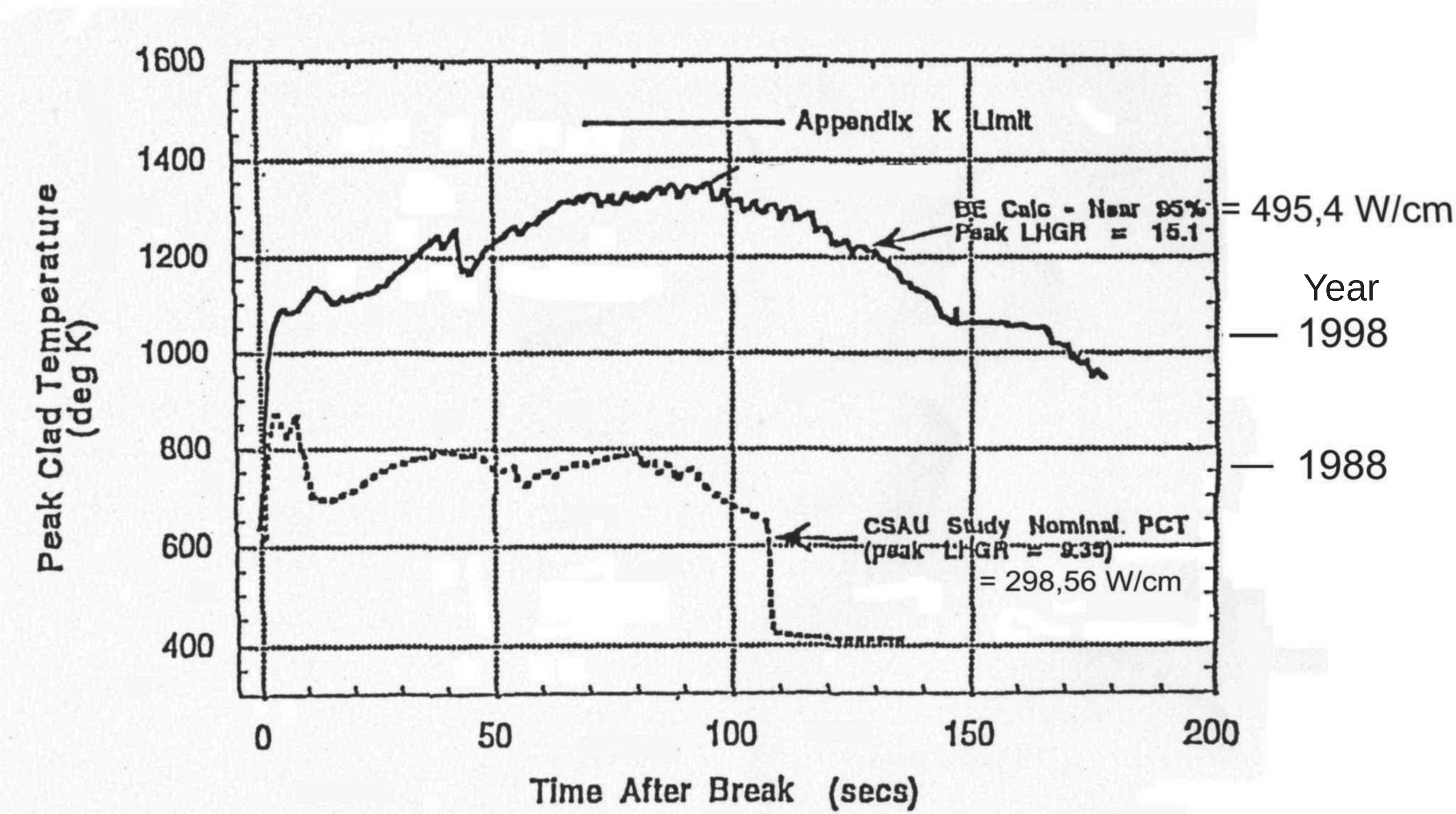
CSAU (Code Scaling Applicability Uncertainty) Method (2)

- **Scaling** considered by identification of several phenomena based on test facilities and on code validation
- Addition of **bias terms** on output uncertainties which are not provided through the analysis
- A **response surface approach** was used in the first demonstrations,
  - Response surface fits the code predictions obtained from selected parameters, and is further used instead of the original computer code
  - Reduces the number of code runs and the cost of analysis
  - Response surfaces are not mandatory within the CSAU framework, other methods for uncertainty quantification may be applied

Sample Best-Estimate Calculation using CSAU Method (USA)

- Peak LHGR = 15.1 (kW/ft) = 495.4 W/cm

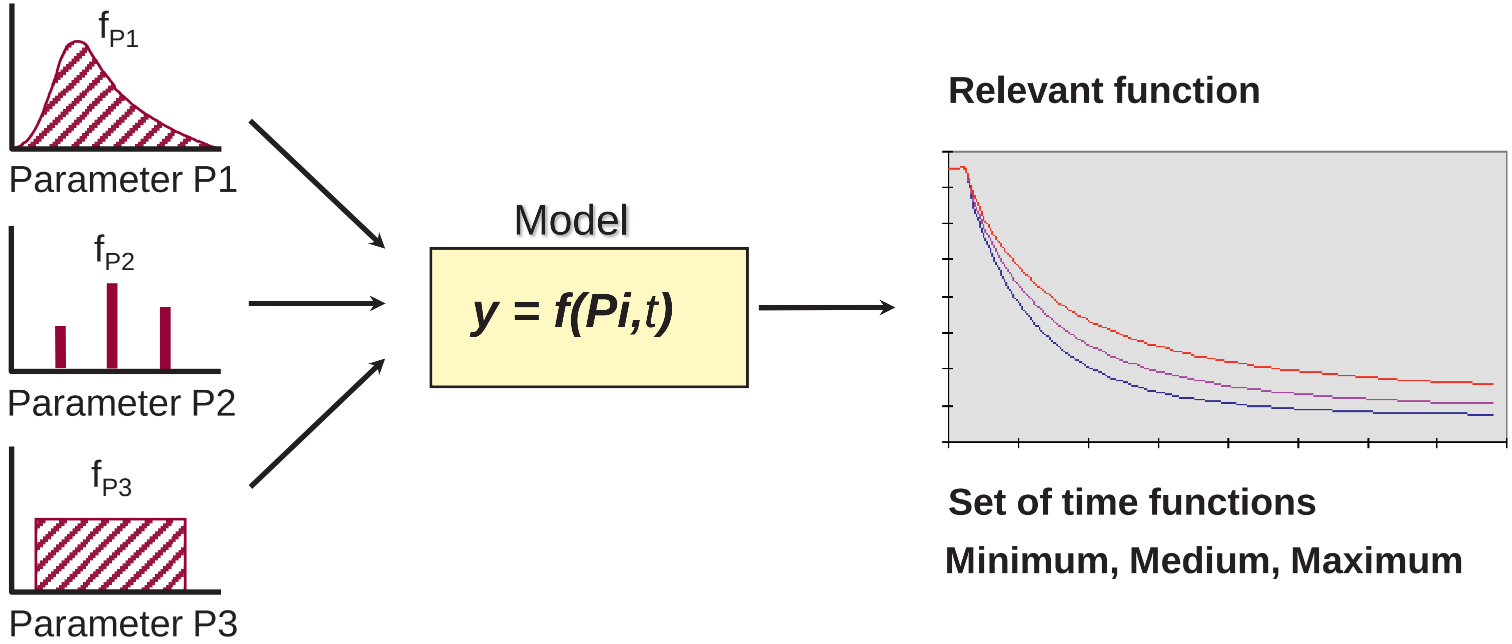
→ Peak Clad Temperature is representative of 95th percentile value



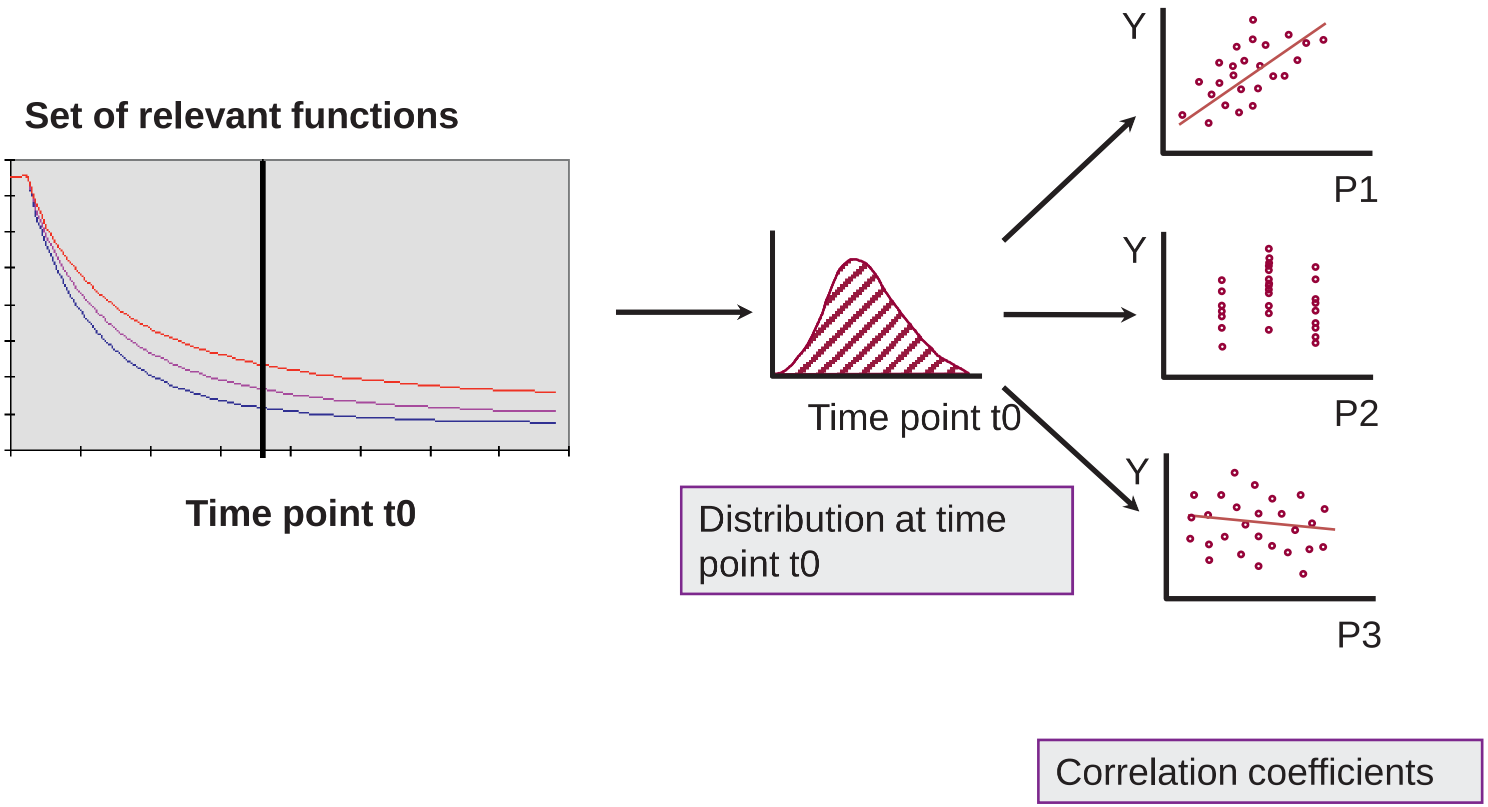
Statistical Methods: GRS, IRSN, AREVA, ASTRUM, GE, KREM, KINS-REM, …

- First proposed by GRS
- **Identify and quantify** all potentially important parameters
- Number of input uncertainties **not** limited (number of code calculations **independent of number of uncertain parameters**)
- Input uncertainties characterised by ranges and probability distributions
- Uncertainty space sampled at random according to the probability distributions
- **Wilks' formula** determines the number of calculations
  - for one-sided 95% confidence limit on the 95th percentile **59 runs** are needed.
  - for two-sided 95%/95% tolerance interval **93 runs** are needed.
- Provides **sensitivity measures** to help prioritise future improvements

Statistical uncertainty analysis



Sensitivity analysis



Data used for quantification of uncertainties

- Results obtained during code validation, envelop results from separate effects and integral tests
  - Relevant and available experimental data should be used
  - Scaling effects considered by large scale experiments, like UPTF
- Data uncertainties from documentation (geometry, bypass flow paths, reactor power, decay heat)
- Fuel data from fabrication tolerances

Number of code calculations - Wilks' formula

- **Independent of number of uncertain parameters!**
- **Dependent on** tolerance limits (or -intervals) for the uncertainty statement of the code results
- **Smallest number of code runs n**
  - upper statistical tolerance limit (one-sided): $1 - \alpha^n \geq \beta$
  - tolerance interval (two-sided): $1 - \alpha^n - n\,(1 - \alpha)\,\alpha^{n-1} \geq \beta$

n	α	$1 - \alpha^n$
10	0,95	0,40
50	0,95	0,92
59	**0,95**	**0,95**
100	0,95	0,99
500	0,95	1,00

n	α	$1 - \alpha^n - n\,(1 - \alpha)\,\alpha^{n-1}$
10	0,95	0,09
50	0,95	0,72
93	**0,95**	**0,95**
100	0,95	0,96
500	0,95	1,00

α % is the desired **probability content** (fractile, percentile, quantile),

β % is the **confidence limit** (taking into account the possible sampling error due to limited number of code calculations)

Number of code calculations - Sequential variation of parameter values, not using Wilks formula

- Selection of maximum, minimum and reference value for each parameter (3 values per parameter)
- **Number of calculations n**
- Without combination of parameters:

  $n = 2p + 1$

  p is number of parameters

- Combination of parameters:

  $\mathbf{n = 3^p}$

  e.g.: $p = 48 \Rightarrow n \approx 8 \times 10^{22}$
  $n = 93 \Rightarrow p_{max} = 4$ (!)   => PIRT process necessary!

Determination of tolerance limits

- A total number of **n code runs** are performed varying simultaneously the values of all uncertain input parameters, according to their distribution
- **For each instant of time** the n values of the considered output parameters are ordered:
  Y(1) <Y(2) … < Y(n-1) < Y(n)
  => "**order statistics**" is used for Wilks' formula
- On the basis of ranking, the **tolerance limits** are obtained **with a confidence level of 95%** by selecting

Number of code runs (samples)	One-sided 95th percentile tolerance limit	One sided 5th percentile tolerance limit	Two-sided tolerance interval
59	Y(n)	Y(1)	Y(1) and Y(n)
93	Y(n-1)	Y(2)	Y(1) and Y(n)
124	Y(n-2)	Y(3)	Y(1) and Y(n)
153	Y(n-3)	Y(4)	Y(2) and Y(n-1)
181	Y(n-4)	Y(5)	Y(3) and Y(n-2)
…	…		…

Comparison with more than 1 acceptance criterion (1)

- A. Wald extended Wilks' concept to several output variables ("Coverage" approach)
- Shortcomings:
  - **Requires** considerably **increased number of code runs**
  - **Depends on** numbering of the output variables, i.e. on the **order in which the output variables are treated** and extreme values are omitted
    => e.g. 1-sided upper tolerance limit:
    1st variable is PCT, **run** with highest PCT eliminated for next output variable,
    2nd variable evaluated without that eliminated run,
    run with highest value of 2nd variable eliminated, etc.

Comparison with more than 1 acceptance criterion (2)

- Slightly **modified concept** proposed:
  - No consideration of joint tolerance limits for the multiple outputs of interest
  - Consideration of the **lower statistical confidence limit** (e.g. of at least 95%) for the **probability of „satisfying all acceptance criteria for all output parameters**" (Clopper-Pearson)

- Basis is that both of the following statements are equivalent:
  - The Wilks' (probability α = 95% and confidence β = 95%) limit for the results is below the regulatory acceptance limit
  - The **lower β = 95% confidence limit** for the probability that the value of the result stays below the regulatory acceptance limit is greater or equal α = 95%.
    The regulatory acceptance limits are incorporated into the probabilistic statements.

Comparison with more than 1 acceptance criterion (3)

- Advantages:
  - In the one-dimensional case of one single output parameter the concept is equivalent to the known concept of one-sided upper tolerance limit
  - Minimum **number of calculation runs is the same for the "multi-dimensional" case, independent of output parameters and criteria involved**, and consequently independent from interrelationships between the output parameters and criteria

Uncertainty analysis provides statements on

- **Uncertainty range** of code results
  - Enables to determine **margin** between upper bound of uncertainty range **to acceptance criterion**

- **Sensitivity measures** about influence of input parameters on calculation results
  - Ranking of parameters as result of the analysis
  - Guides further code development
  - Prioritises experimental investigations

Sensitivity measures

- correlation coefficient
  ⇒ measure of linear relations of one parameter to the result
- partial correlation coefficient
  ⇒ measure of linear relations of one parameter to result after elimination of linear effects of other parameters
  (not recommended, is ratio of parts of variability rather than fraction of variability, may show higher measure at low influence)
- standardised regression coefficient
  ⇒ linear relation of one input parameter to variability of result after elimination of linear effects of other parameter variabilities
- rank transformation (linear and monotonic dependence of ranks)
- correlation ratios
  ⇒ not restricted to linear and monotonous relations

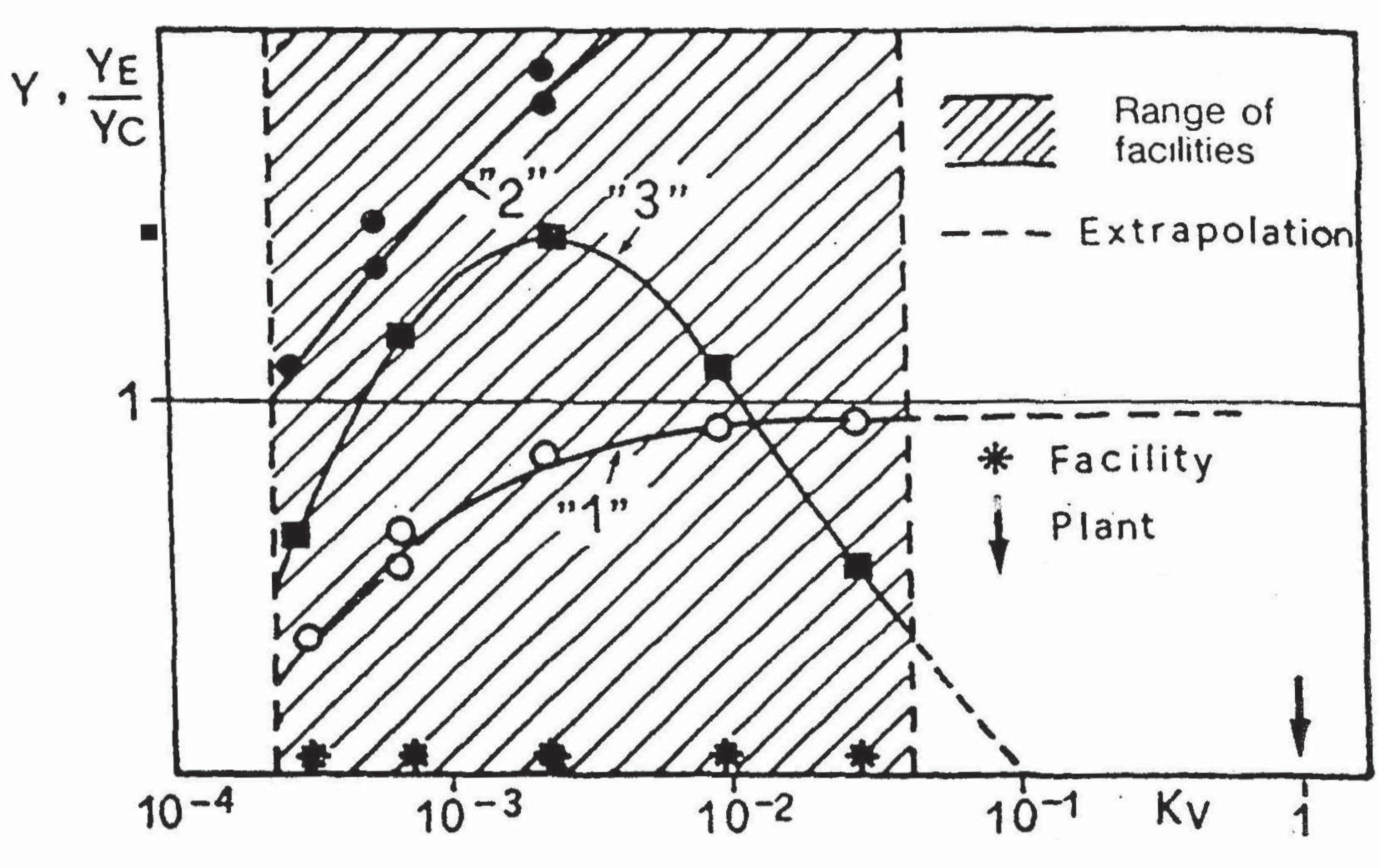
Support Programmes SUSA (GRS), SUNSET (IRSN), SNAP/DAKOTA (NRC)

- Provides a choice of statistical tools to be applied during the uncertainty and sensitivity analysis
- Supports analyses during the different working steps
- Supports evaluation of results

University Pisa Method - Uncertainty Methodology based on Accuracy Extrapolation (UMAE)

- No consideration of input uncertainties
- Quantitative **determination of accuracy** of code calculations by means of integral experiments based on Fast Fourier Transform (FFT) for the investigated plant scenario
- Calculation of final uncertainty by **extrapolation of accuracy evaluated in predicting integral experiments** to full scale reactor plant
- **Suitably scaled facilities** and **relevant data** from integral experiments **must exist!**

Uncertainty Methodology based on Accuracy Extrapolation (UMAE)



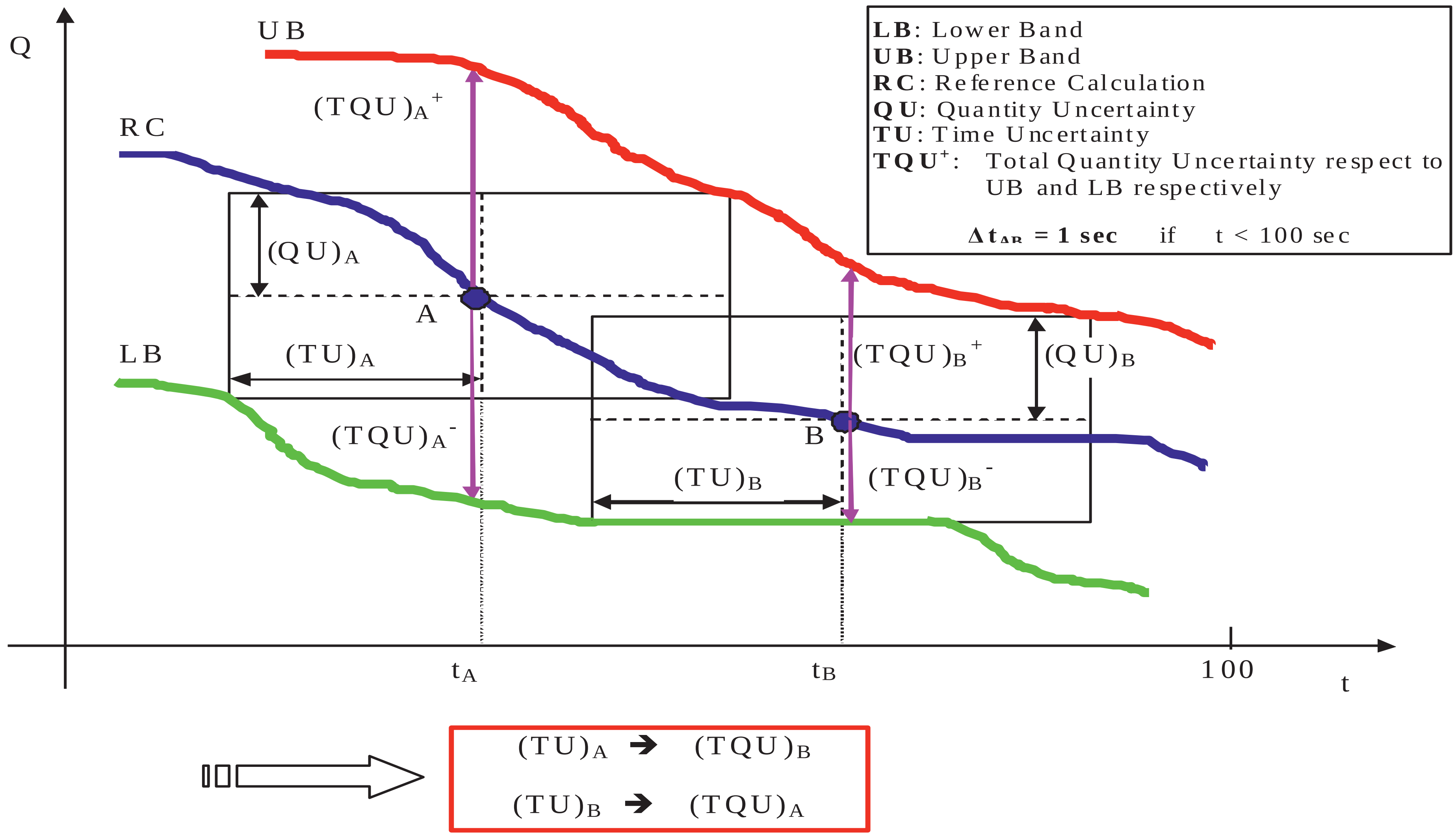
University Pisa Method - Uncertainty Methodology based on Accuracy Extrapolation (UMAE)

- Code modelling of the integral experiment data must **satisfy criteria for prediction** of relevant thermal-hydraulic aspects and accuracy
- **Same (qualified) noding** used for plant calculation
- Accuracy of calculations for integral experiments extrapolated to plant; formula allows for effects of scale, most likely to be when extrapolation is small
- **No sensitivity information** between input and output parameters without additional specific calculations, beyond the scope of UMAE

Code with capability of Internal Assessment of Uncertainty (CIAU)

- Each plant state is characterized by the value of **6 relevant quantities** (i.e. a hypercube) **and** by the value of **time** since transient start
- An uncertainty can be assigned to each plant state
- For PWRs the 6 quantities are:
  - Upper plenum pressure
  - Primary loop mass inventory including pressurizer
  - Steam generator secondary side pressure
  - Cladding surface temperature at 2/3 of core active height (from bottom of active fuel)
  - Core power
  - Steam generator downcomer collapsed liquid level (the largest value of different SGs)
- The value of uncertainty – corresponding to each edge of the rectangle – can be defined in probabilistic terms => this shall satisfy the requirement of a 95% probability level according to US 10 CFR 50 and Regulatory Guide 1.157
- This **time and resource consuming process** has been performed and is available only at University Pisa for RELAP5 and CATHARE codes up to now

Quantity Uncertainty, Time Uncertainty and Total Quantity Uncertainty of the CIAU method of University Pisa



Comparison of relevant features of uncertainty methods (1)

Feature	CSAU demo	Statistical/ GRS	UMAE/CIAU
Determination of uncertain input parameters and of input uncertainty ranges	Experts	Experts	Differences between experimental and used input data constitute sources for uncertainty of code models
Selection of uncertain parameter values within the determined range for code calculations	Experts	Random selection	Not necessary
Support of identification and ranking of main parameter and modelling uncertainties (PIRT)	Yes	No (optional)	No
Accounting for state of knowledge of uncertain parameters (distribution of input uncertainties)	Yes	Yes	No

Comparison of relevant features of uncertainty methods (2)

Feature	CSAU demo	Statistical/ GRS	UMAE/CIAU
Probabilistic uncertainty statement	Yes	Yes	Yes
Statistical rigour	No	Yes	No
Knowledge of code specifics may reduce resources necessary for the analysis	Yes	No	No
Number of code runs independent of number of input and output parameters	No	Yes	Yes
Typical number of code runs	LB: 8 SB: 34	59 PWR: 93-300 LOFT: 59-150 LSTF: 59-100	Not applicable, Roughly 20

Comparison of relevant features of uncertainty methods (3)

Feature	CSAU demo	Statistical/ GRS	UMAE/CIAU
Number of uncertain input parameters	LB: 7 (+5) SB: 8	LOFT: 13-64 PWR: 17-55 LSTF: 25-48	Not applicable
Quantitative information about influence of a limited number of code runs	No	Yes	No
Use of response surface to approximate the result	Yes	No	No
Use of biases on results	Yes	No	For other than model uncertainties

Comparison of relevant features of uncertainty methods (4)

Feature	CSAU demo	Statistical/ GRS	UMAE/CIAU
Continuous valued output parameters	No	Yes	Yes
Sensitivity measures of input parameters on output parameters	No	Yes	No

Best estimate analysis including uncertainty analysis

Used in licensing up to now in:

- USA
- Netherlands
- Brazil (Siemens, CIAU)
- Korea
- Lithuania
- France
- Spain
- Belgium
- China
- Taiwan
- Argentina (CIAU)
- Great Britain

Significant activities for use in licensing in:

- Canada
- Czech Republic
- Hungary
- Japan
- Russia
- Slovak Republic
- Ukraine
- Germany

Conclusions

- Uncertainty analysis is becoming common practice world-wide, mostly statistical method used
- Basis for applications of statistical uncertainty evaluation methods is the GRS-method
- Extrapolation of output uncertainties proposed by University Pisa
- Comparison of applications of existing uncertainty methods have been performed in the frame of OECD/ CSNI Programmes (UMS and BEMUSE)
  - Differences of results may come from
    - Different methods
    - For UMAE/ CIAU different number of experiments for codes CATHARE and RELAP
    - For statistical methods due to different input uncertainties, their ranges and distributions as well as reference calculations
- Application of statistical methods: Further activity will be focussed on specific procedures to determine input uncertainties of code models
  => OECD PREMIUM (**P**ost BEMUSE **RE**flood **M**odels **I**nput **U**ncertainty **M**ethods) Project
- Determination of input uncertainties as well as quality of reference calculation is most important for uncertainty analysis

IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Information Synthesis in Uncertainty Studies : Application to the Analysis of the BEMUSE Results

J. Baccou, E. Chojnacki and S. Destercke

Content

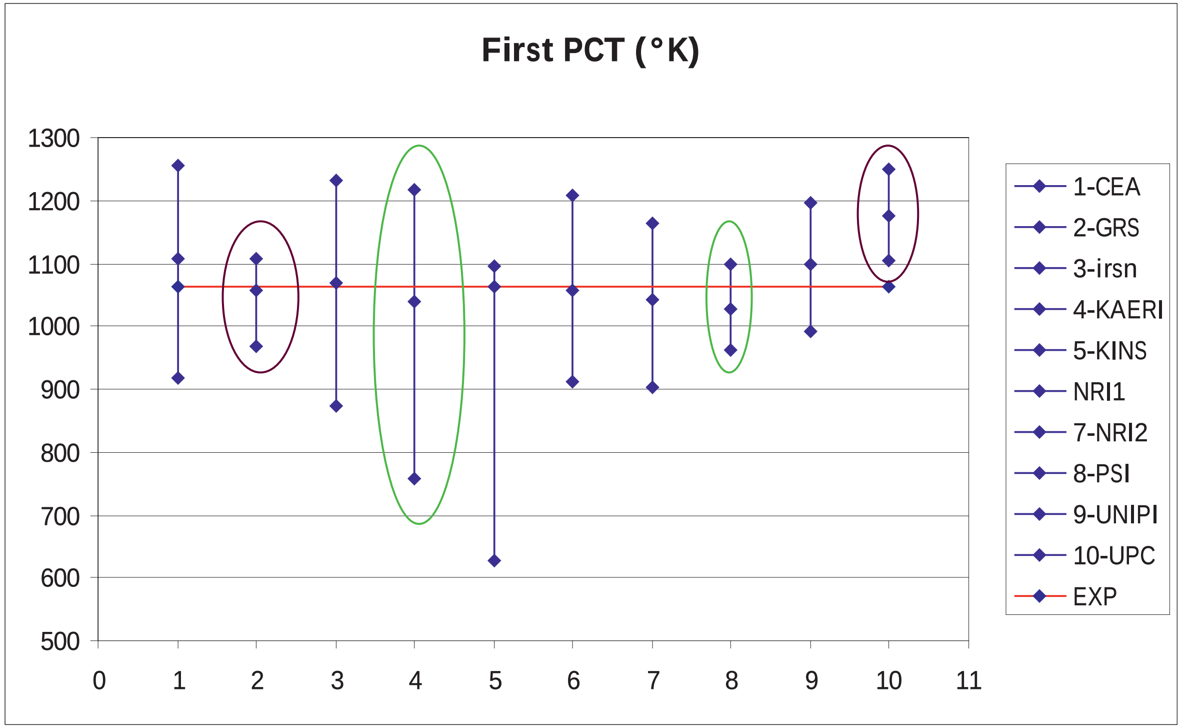
1) Introduction on information synthesis

2) Construction of the method

3) Application in the frame of the BEMUSE program

4) Conclusion

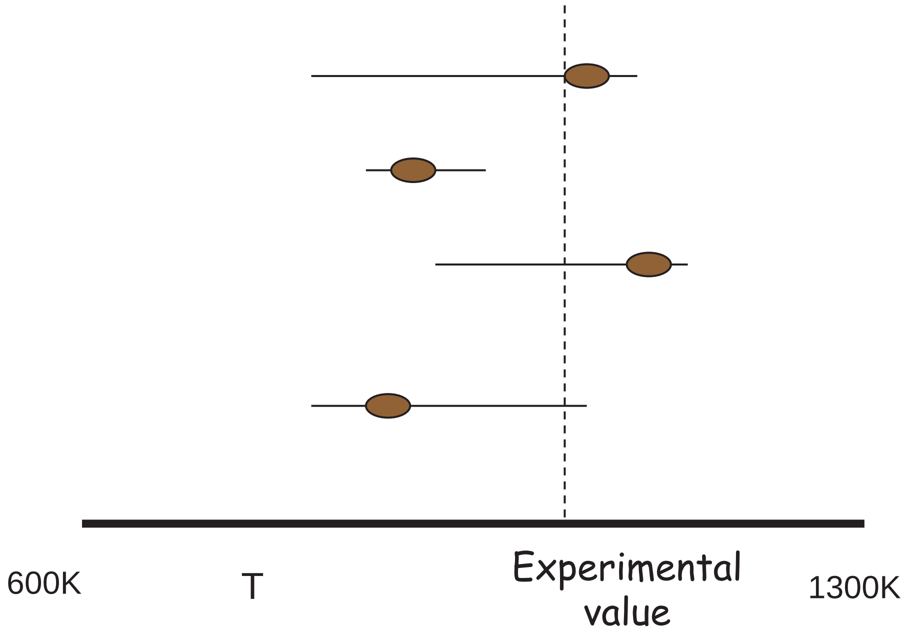
1) The problem of information synthesis



Large/Small uncertainty bands

Discrepancy BE/Exp Value

2) Construction of the method



1) Modelling of information provided by each participant: choose a mathematical framework to represent the available information

2) Evaluation of the quality of the information: define and compute numerical criteria to take into account the precision of the information and its coherence with observed reference values .

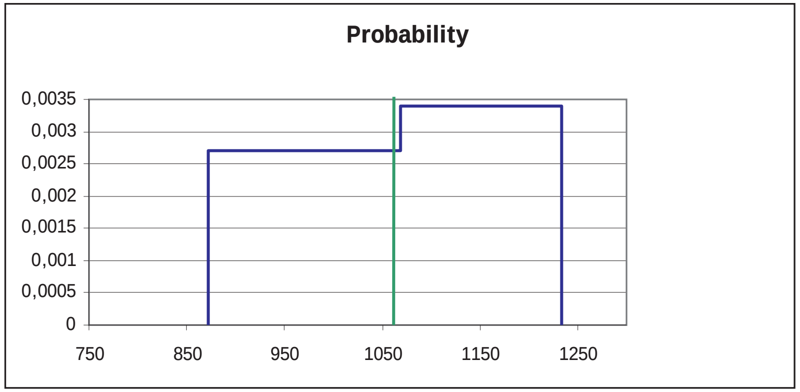
3) Information fusion: it implies the definition and the application of fusion operators to build a summary of all information provided by the sources.

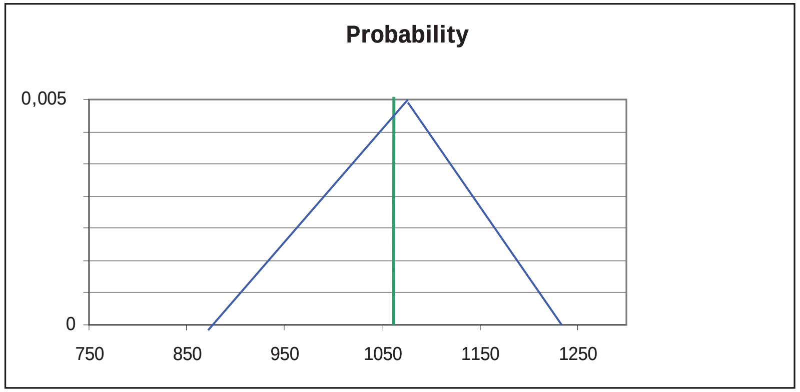
2) Information modelling

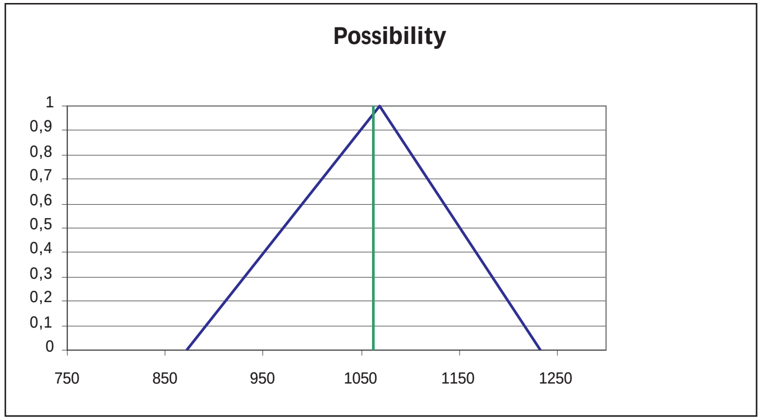
Two examples of knowledge model on an example:

PCT given by a participant, [872,1233], BE=1069, Exp= 1062

Probability







Possibility:

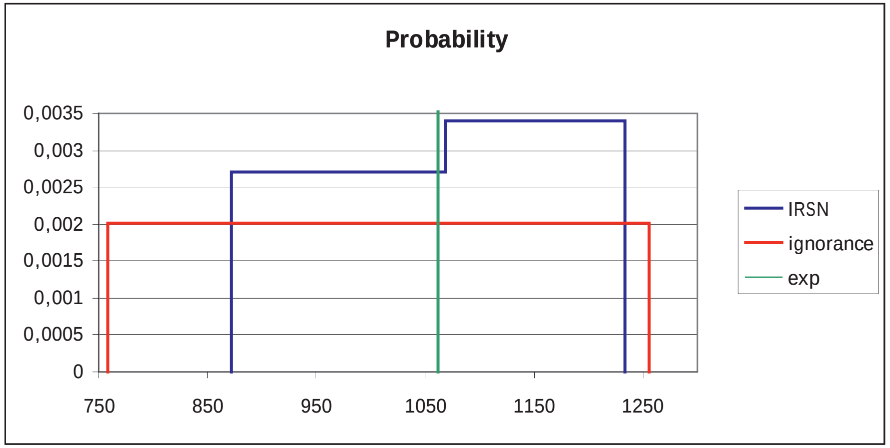
Partial probabilistic model, more adapted to the available state of knowledge (interval + BE value)

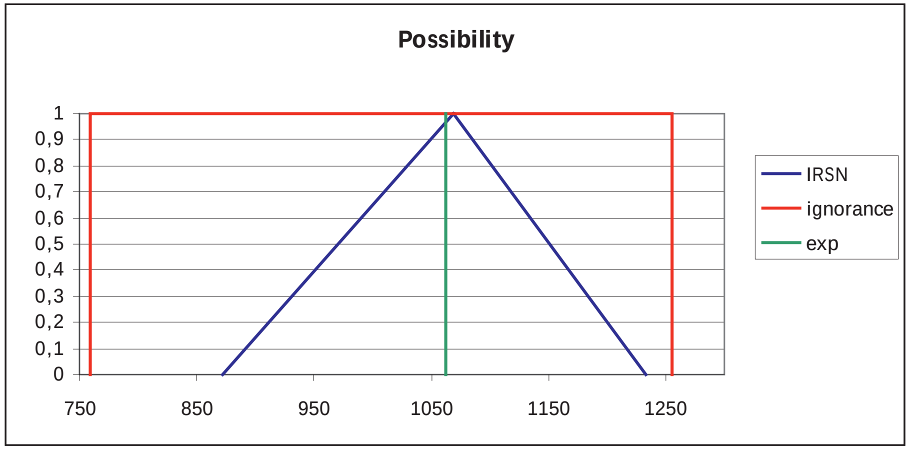
2) Information evaluation

Two criteria : Informativeness & Calibration

Their computation depends on the mathematical framework (probability or possibility)

- Informativeness: : it measures the precision of the information. The more precise a source is, the more useful it is





$$I(p,u) = \sum_{i=1}^{B} p_i \log\left(\frac{p_i}{u_i}\right)$$ (relative entropy)

$$I(\pi, s) = \frac{|\pi_{ign}| - |\pi_s|}{|\pi_{ign}|} \qquad |\pi_s| = \int_{q_l}^{q_u} \pi(x)dx$$

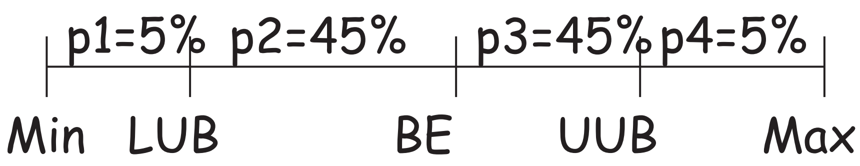
if information on N variables, global informativeness =(weighted) mean of informativeness scores over all these variables.

Same concept : small uncertainty range

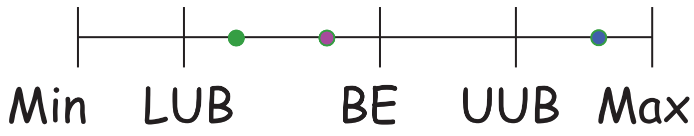
- **Calibration:** it measures the coherence between information provided by a participant and the experimentally observed value
  - Probability:

For each output of interest: Min, LUB, BE, UUB, Max

Theoretical distribution



Experimental distribution taken into account all the output variables



0%	100%	0%	0%
0%	50%	0%	50%
0%	66.7%	0%	33.3%
		...	
r1	**r2**	**r3**	**r4**

$$I(r,p) = \sum_{i=1}^{B} r_i \log\left(\frac{r_i}{p_i}\right)$$

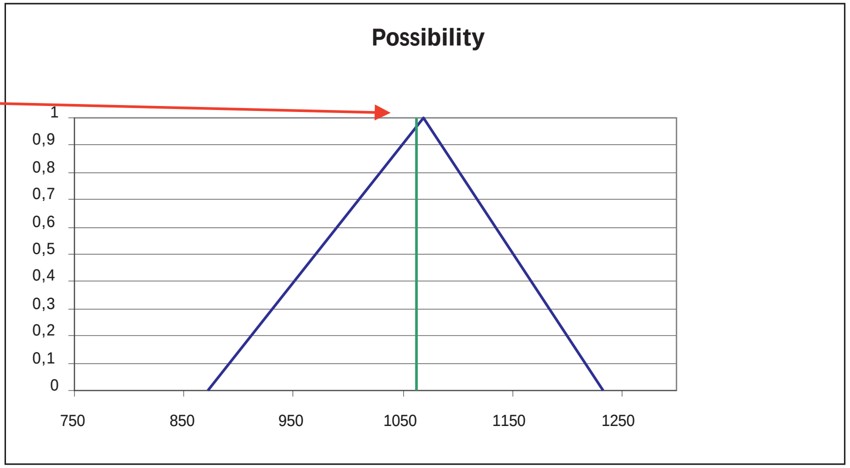
$$Cal_p(s) = 1 - \chi^2_{B-1}\left(2 * N * I(r,p)\right)$$

Output variables « well distributed around nominal (BE) value »

- Possibility:

  For each output of interest:

$$Cal(\pi, \exp) = \pi(\exp)$$



Calibration averages the distance between observed values and the expected ones

Different concept : output variables « close to » the nominal (BE) values

- Final score: product of both calibration and informativeness

2) Information fusion

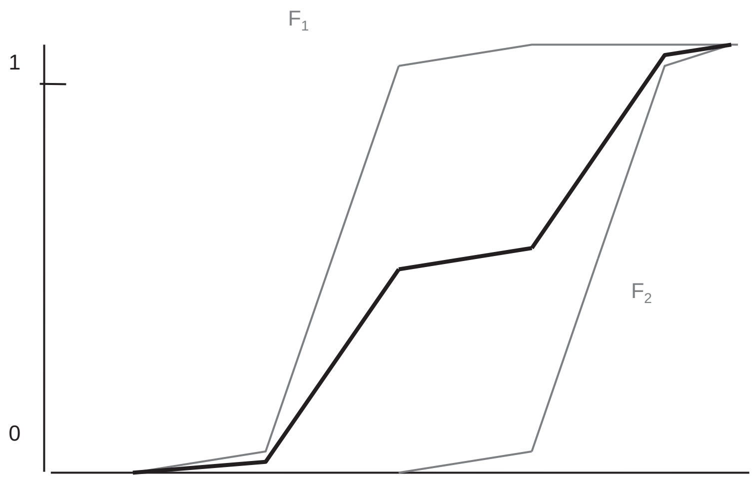
- Three main fusion operators :

Disjunctive (→ union) ,
All information given by each source

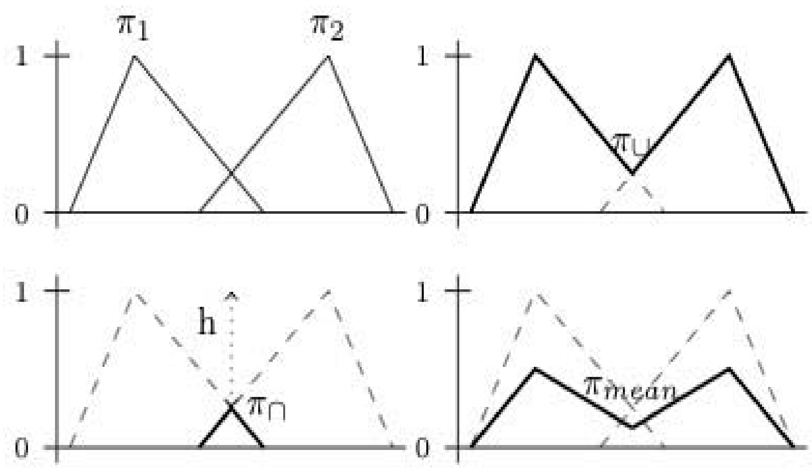
Conjunctive (→ intersection) ,
Information common to all sources

Weighted average ,
Average the information given by each source

Probabilistic framework



Possibilistic framework



3) Application in the frame of the BEMUSE program

BEMUSE-phase 3 (LOFT experiment): evaluation results (IRSN SUNSET software)

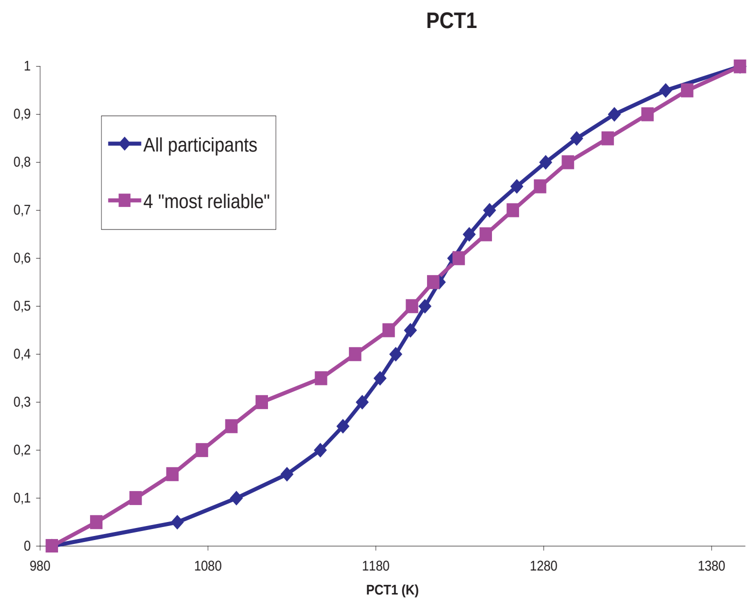
Participants	Infor. Proba	Calib. Proba	*Global Proba*	Infor. Poss	Calib. Poss	*Global Poss*
CEA	8	5	*6*	8	7	*7*
IRSN	5	2	*2*	6	1	*1*
GRS	4	1	*1*	3	6	*6*
NRI2	6	8	*8*	4	2	*2*
KAERI	9	5	*7*	9	8	*8*
PSI	1	10	*10*	1	10	*10*
KINS	3	5	*5*	7	3	*3*
NRI1	7	2	*3*	5	5	*4*
UNIPI	10	2	*4*	10	4	*5*
UPC	2	9	*9*	2	9	*9*

Good agreement with the direct analysis and also between formal methods

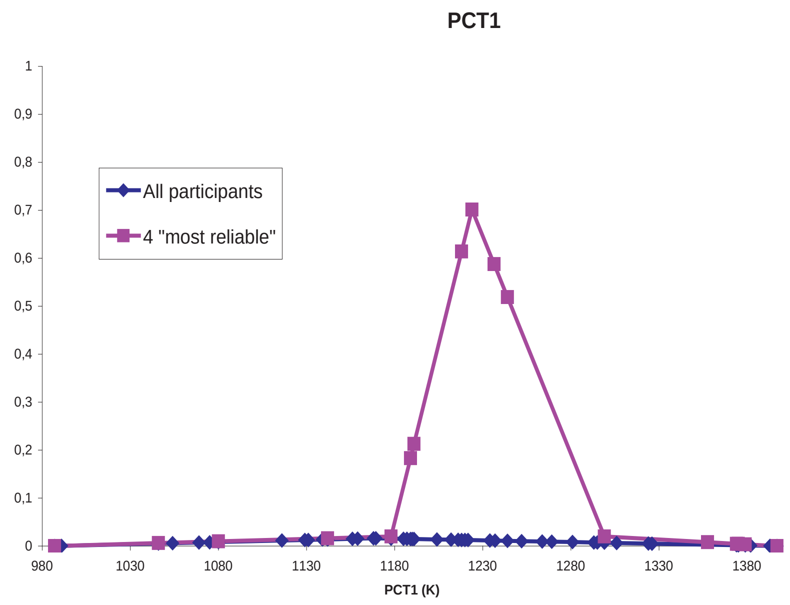
BEMUSE-phase 5 (Zion experiment): fusion results (IRSN SUNSET software)

With respect to the mathematical modelling

Probabilistic framework



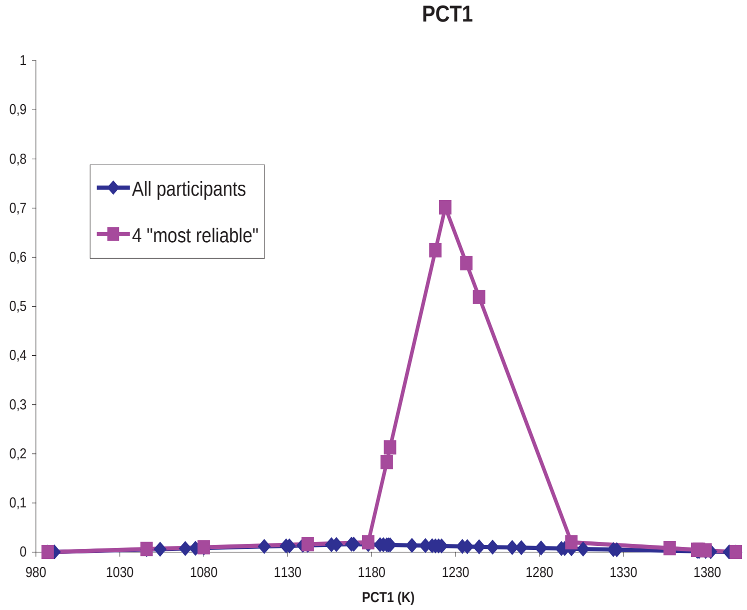
Possibilistic framework

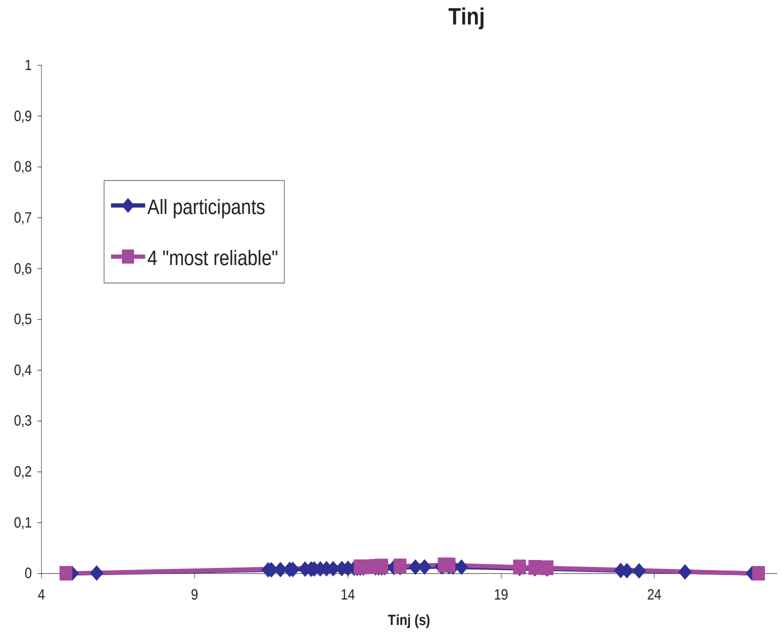


Probability distributions provided by the four most reliable sources of the LOFT benchmark close to the distribution aggregated from all participants

-Results provided by all participants highly conflicting due to several uncertainty ranges that do not overlap.
- Considering the four most reliable sources strongly increases the coherence of the results.

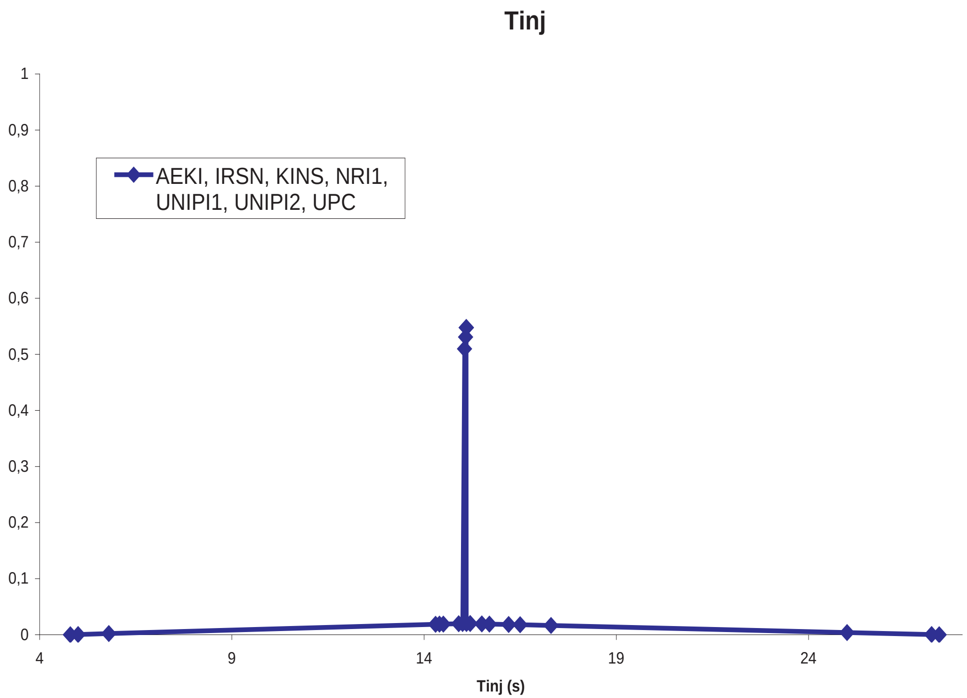
With respect to the scalar output variable

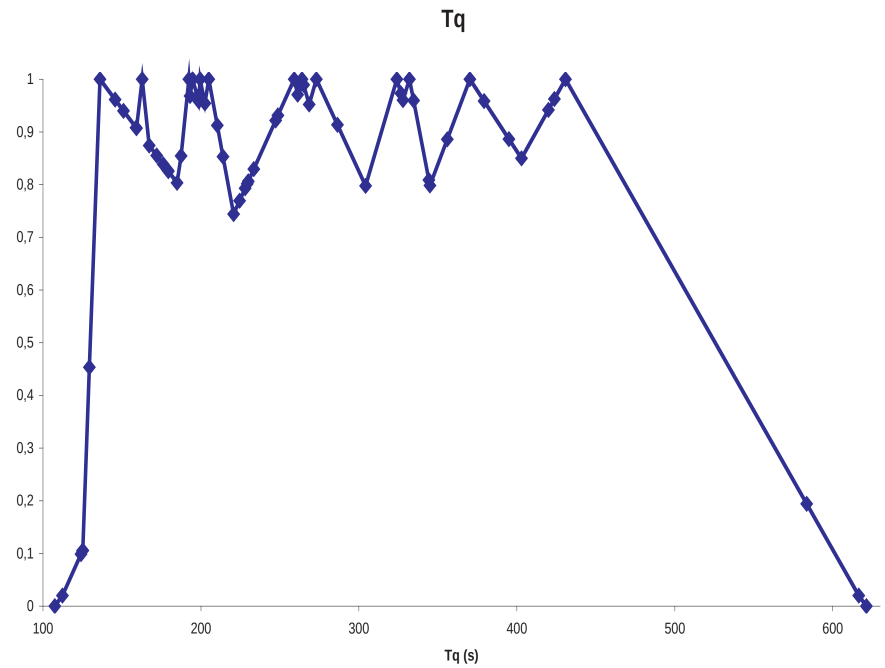




Participants are more conflicting as for time variables

A more reliable synthesis for time variables is the one based on the union of information provided by each participant



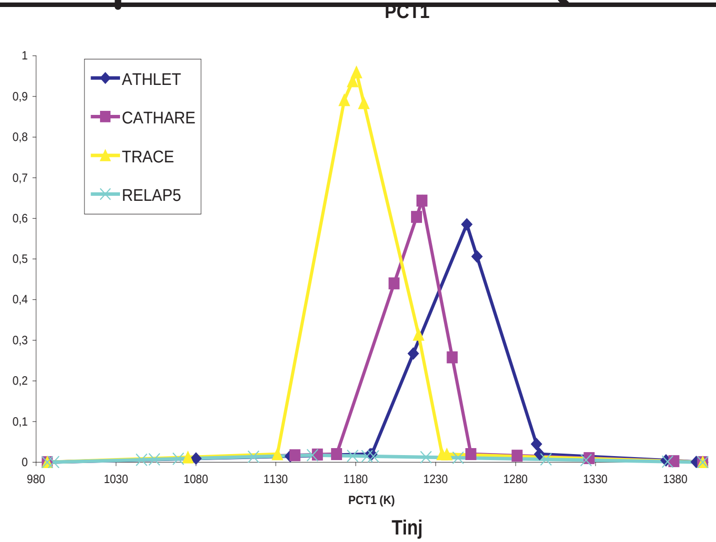


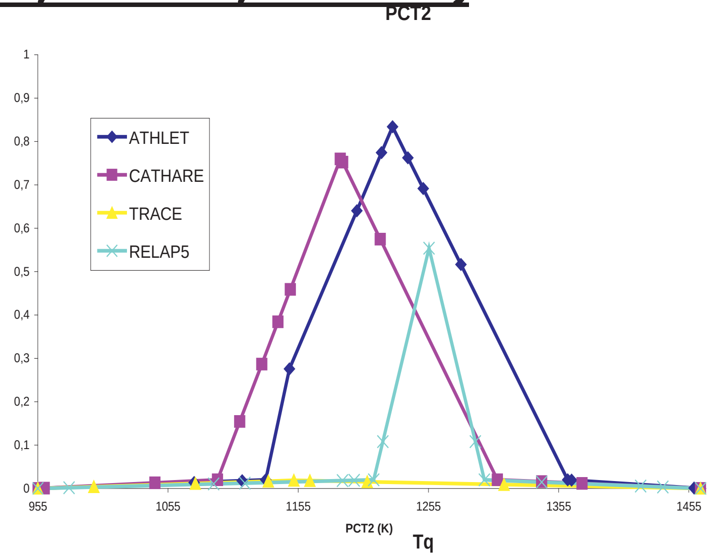
Accumulator injection time:

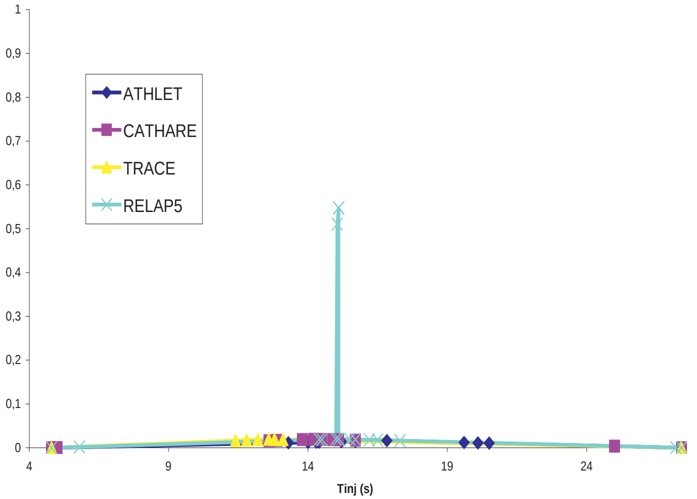
-Narrow uncertainty margins indicating that the uncertainties which have been taken into account don't impact this output variable.

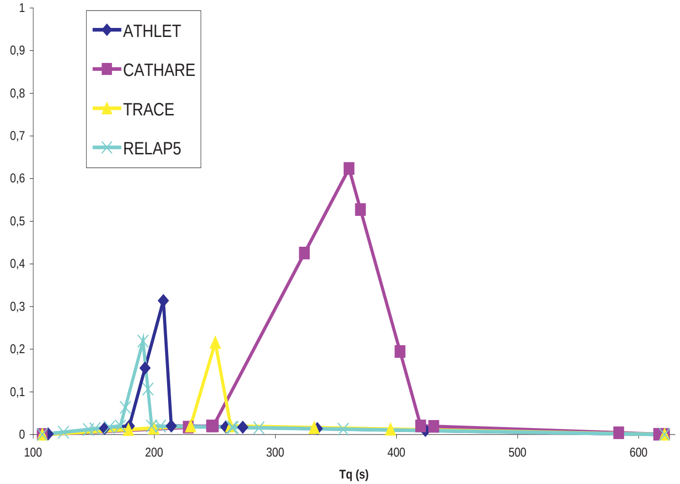
- Coherent Subgroups of participants (AEKI, IRSN, KINS, NRI1, UNIPI1, UNIPI2, UPC)

With respect to used code (ATHLET, CATHARE, TRACE, RELAP5)









- Code effect not negligible in the estimation of ref. calc. and uncertainty margins

- Uncertainty estimation more coherent for « temperature » than for « time » (due to dispersion of ref. calc. and narrow uncertainty margins)

4) Conclusion

- Available information: LUB, BE (reference) value, UUB
- 3 steps:
  - Information modelling: possibility framework more adapted to the state of knowledge,
  - Information evaluation: informativeness&Calibration (depends on the mathematical framework for information modelling)
  - Information fusion: large choice of fusion operators

    ⟶ Identification of concordant/discordant participants
- Synthesis of the BEMUSE results:
  - All participants: information highly conflicting for the four scalar outputs of interest (first and second peak cladding temperature, injection and quenching time).
  - Sub-groups of participants (identified with respect to the quality of the provided information on the LOFT benchmark and to the used code):
    - More coherent results related to temperature but not to time variables
    - Code effect not negligible

IRSN

11/16/2011

Generic Application of Wilks' Tolerance Limit Evaluation Approach to Nuclear Safety

OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations

In Seob Hong

D.Y. Oh and I.G. Kim

Korea Institute of Nuclear Safety

Outline

- Background
- Wilks' Integral Formula Set
- Wilks' Combinatorial Formula Set
- Numerical Validation
- Discussion
- Wrapup

Background

- **BE (Best Estimate):** An idea to determine 'realistic operating limits' which came from the 'rare-event' concept. Represents the state-of-knowledge thus always contains a certain level of uncertainty.

- **BEPU (BE Plus Uncertainty):** BE analyses should be followed by uncertainty analyses (UA) to derive meaningful conclusions.

- **Tolerance Limit Approach (by Wilks) and Response Surface (RS) method are popular methods to determine the uncertainty tolerance in the BEPU framework.**

- **Wilks' formula set was suggested to the nuclear safety analysis by GRS (Gesellschaft für Anlagen- und Reaktorsicherheit).**

- **Tolerance limit**
  - **Uncertainties come from various sources and are cased and propagated into certain (input and output) parameters, which can be expressed in terms of (continuous or discrete) parameter ranges. A certain (output) parameter has a probability density function (PDF), and a range of interest (tolerance limit) can be expressed using:**

$$\alpha = \int_{x_L}^{x_U} f(x)dx\ , \quad \text{or its complementary,} \quad i.e.,\ 1-\alpha$$

α : *tolerance limit in terms of cumulative probability interval,*

x_U *and* x_L : *upper and lower tolerance limits for variable* x.

  - **How to express tolerance limit (cumulative probability interval) conventionally?**

$$\alpha^{th}\ \ percentile\ : \alpha \times 100^{th}\ \ldots$$

- **How is an $\alpha - relevant$ event relates to a probability statement?**

$$Pr(a \ \ certain \ \ \alpha - relavant \ \ event \ \ to \ \ occur) \geq \beta, \ \beta \times 100\% \ \ confidence \ \ level$$

Common practice: 95th percentile with 95% confidence level.

- **What happens if we formulate the above probability statement only in terms of α in lieu of f(x)?**
  - **It becomes distribution (f(x)) free.**
  - **Wilks' formula set deals with only cumulative probability thus it is referred to as 'Distribution-free tolerance limit approach'.**

- **BEMUSE projects have been actively conducted under the lead of OECD/NEA, CSNI.**

- **Queries observed in the BEMUSE Projects**
  - **Tolerance limit evaluation approach has widely been adopted among many organizations. But there are still open topics:**

    Examples)

    Adoption of different code runs for the maximum search above the 95th/95% percentile/confidence level:

    - **59 code runs (one-sided 1st order)**
    - **100 code runs**
    - **124code runs (one-sided 3rd order)**

    Understanding of two-sided approach:

    - **many people believe that the two-sided 1st order can be treated exactly the same as the one-sided 2nd order approach.**

Q1. Which number is more appropriate to meet the 95th / 95 % requirement?

Q2. Can we all agree on the 95th / 95 % practice from the safety perspective?

Wilks' Integral Formula Set (original)

- **Wilks suggested one- and two-sided formula in the form of PDFs for r^th order statistics: [S.S. Wilks, "Determination of Sample Sizes for Setting Tolerance Limits" *in* 1941]**

$$f(u) = \frac{\Gamma(n+1)}{\Gamma(r)\Gamma(n-r+1)} u^{r-1}(1-u)^{n-r} : single\ event\ \ pdf : one\text{-}sided$$

$$f(u,v) = \frac{\Gamma(n+1)}{\Gamma^2(r)\Gamma(n-2r+1)} u^{r-1}v^{r-1}(1-u-v)^{n-2r} :\ joint\ event\ \ pdf : two\text{-}sided$$

- **He used numerical integrations on the (r^th order) PDFs to estimate the cumulative related tolerance limit.**

$$F(u) = \int_a^b f(u)du,\ F(u,v) = \int_d^e \int_a^b f(u,v)dudv$$

- **It may not be very trivial to understand and even reproducing the Wilks' formula set, which is used in the nuclear industry.**

- **The above PDFs are differential forms of series of Binomial Probability Mass Functions (PMFs):**

$$f(\alpha)=\frac{d}{d\alpha}F_{\alpha_k}(k;n,\alpha)=\frac{d}{d\alpha}\sum f_{\alpha_k}(k;n,\alpha),\ where\ f(k;n,\alpha)={}_nC_k\,\alpha^k(1-\alpha)^{n-k}$$

- **Therefore, the process of differentiation and then integration of a PMF can be reduced to a direct derivation of the PMF.**

Wilks' Combinatorial Formula Set

The Present One- and Two- sided Wilks' Formula Set in Nuclear Industry

- **One-sided 1st Order Formula in Nuclear Industry:**

$$1-\alpha^{n} \geq \beta \ (complementary)$$

$$\sum_{k=0}^{n-1} {}_{n}C_{k}\alpha^{k}(1-\alpha)^{n-k} \geq \beta \ (series) \ by \ A. \ Guba$$

- **Two-sided 1st Order Formula in Nuclear Industry:**

$$1-\alpha^{n}-n(1-\alpha)\alpha^{n-1} \geq \beta \ (complementary)$$

$$\sum_{k=0}^{n-2} {}_{n}C_{k}\alpha^{k}(1-\alpha)^{n-k} \geq \beta \ (series) \ by \ A. \ Guba$$

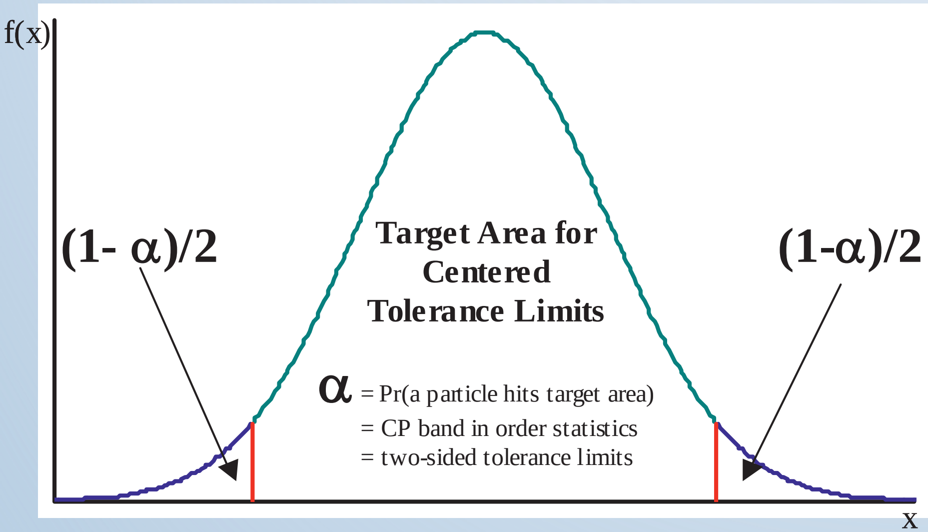
- **One-sided 2nd Order Formula in Nuclear Industry:**
  - ✓ **The same as the two-sided 1st order formula (?)**

- **Conventional Understanding of the Present Formula Set:**
  - a minimum number of code runs can be determined by the above equation set for a given 'tolerance limit'/'confidence level' set.
  - For example, for a 95th percentile, by
    - ✓ One-sided formula: we expect that n=59 code runs will produce the maximum value of an output parameter of interest, which will be located at larger than the 95th percentile with 95% confidence. (This is used to estimate maximum limit.)
    - ✓ Two-sided formula: we expect that n=93 code runs will produce the maximum and minimum values of an output parameter of interest, which will be located outside of the 95th percentile with 95% confidence. (This is used for bounding study.)

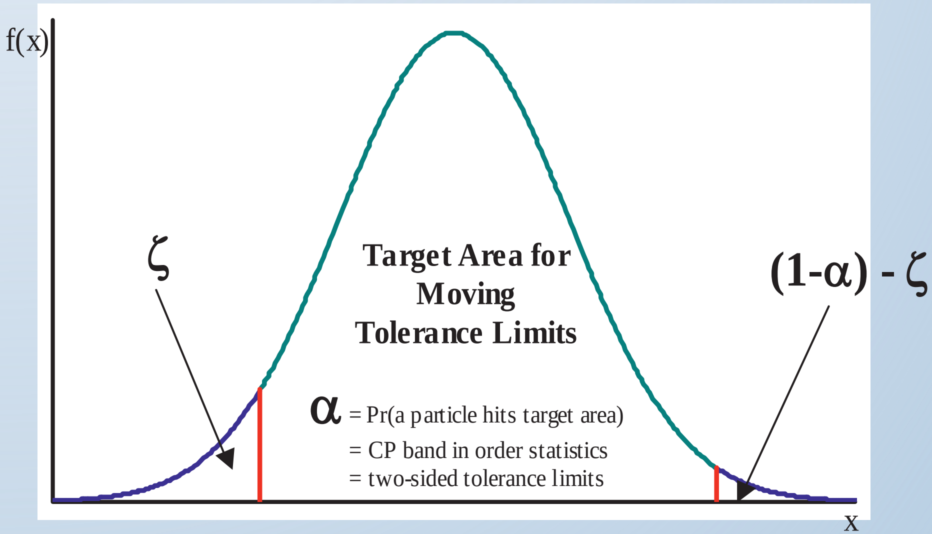
- **It was confirmed that the one-sided approach is fine but may not for the two-sided approach.**

Comparison of Suggestion and Present Wilks' Two-sided Approaches

Two-sided Suggested



Present Two-sided by GRS



Suggestion is for equally truncated bounding study !

- ***Note: A small difference in the percentile location may end in a big difference in the actual parameter output (more important in case of highly skewed PDF at longer tailed side).***

The Suggested One- and Two- sided Wilks' Formula Set for Nuclear Industry

- **One-sided p-th Order Formula :**

$$1-\sum_{k=n-p+1}^{n} {}_nC_k\alpha^k(1-\alpha)^{n-k} \geq \beta \ (complementary)$$

  - **The same as the** $\sum_{k=0}^{n-p} {}_nC_k\alpha^k(1-\alpha)^{n-k} \geq \beta \ (series) \ by \ A.\ Guba$

- **Two-sided 1-st Order Formula : Not the same as the present formula (Centered percentile)**

$$Pr\left\{(CP(x_m) \leq \frac{1-\alpha}{2}) \cap (CP(x_M) \geq \frac{1+\alpha}{2}); the\ 1st\ order\right\} = 1+\alpha^n - 2\alpha^n \sum_{k=0}^{n} {}_nC_k\left(\frac{1-\alpha}{2\alpha}\right)^k \geq \beta$$

- **Two-sided 2-nd Order Formula : Newly introduced here (Centered percentile)**

$$Pr\left\{(CP(x_m) \leq \frac{1-\alpha}{2}) \cap (CP(x_M) \geq \frac{1+\alpha}{2}); the\ 2nd\ order\right\} =$$

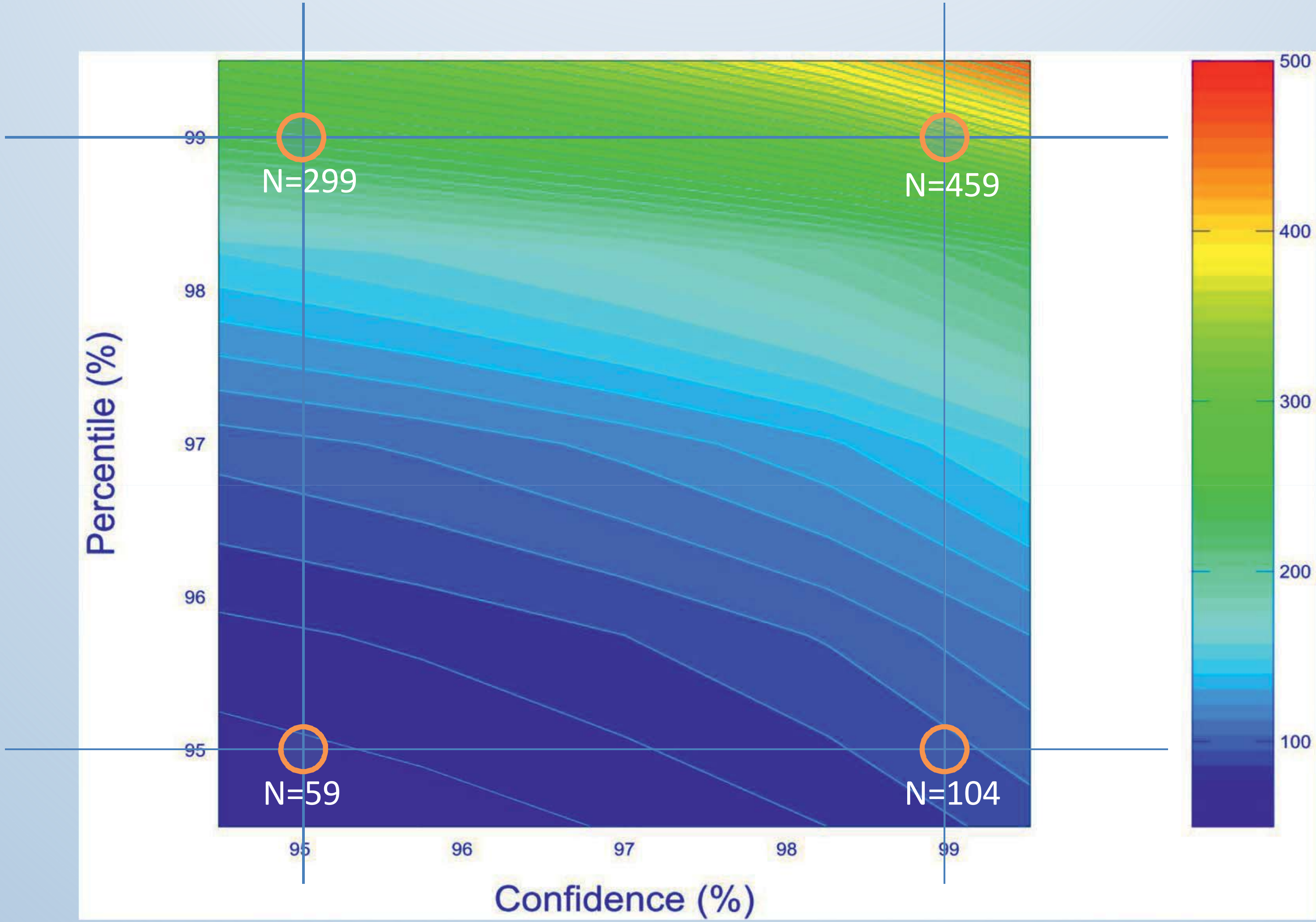
$$2\sum_{k=5}^{n}\sum_{r=2}^{\left[\frac{k-1}{2}\right]} {}_nC_{k-r}\left(\frac{1-\alpha}{2}\right)^r {}_{n-k+r}C_r\left(\frac{1-\alpha}{2}\right)^{k-r}\alpha^{n-k} + \sum_{r=2}^{\left[\frac{n}{2}\right]} {}_nC_r\left(\frac{1-\alpha}{2}\right)^r {}_{n-r}C_r\left(\frac{1-\alpha}{2}\right)^r \alpha^{n-2r} \geq \beta$$

Numerical Validation

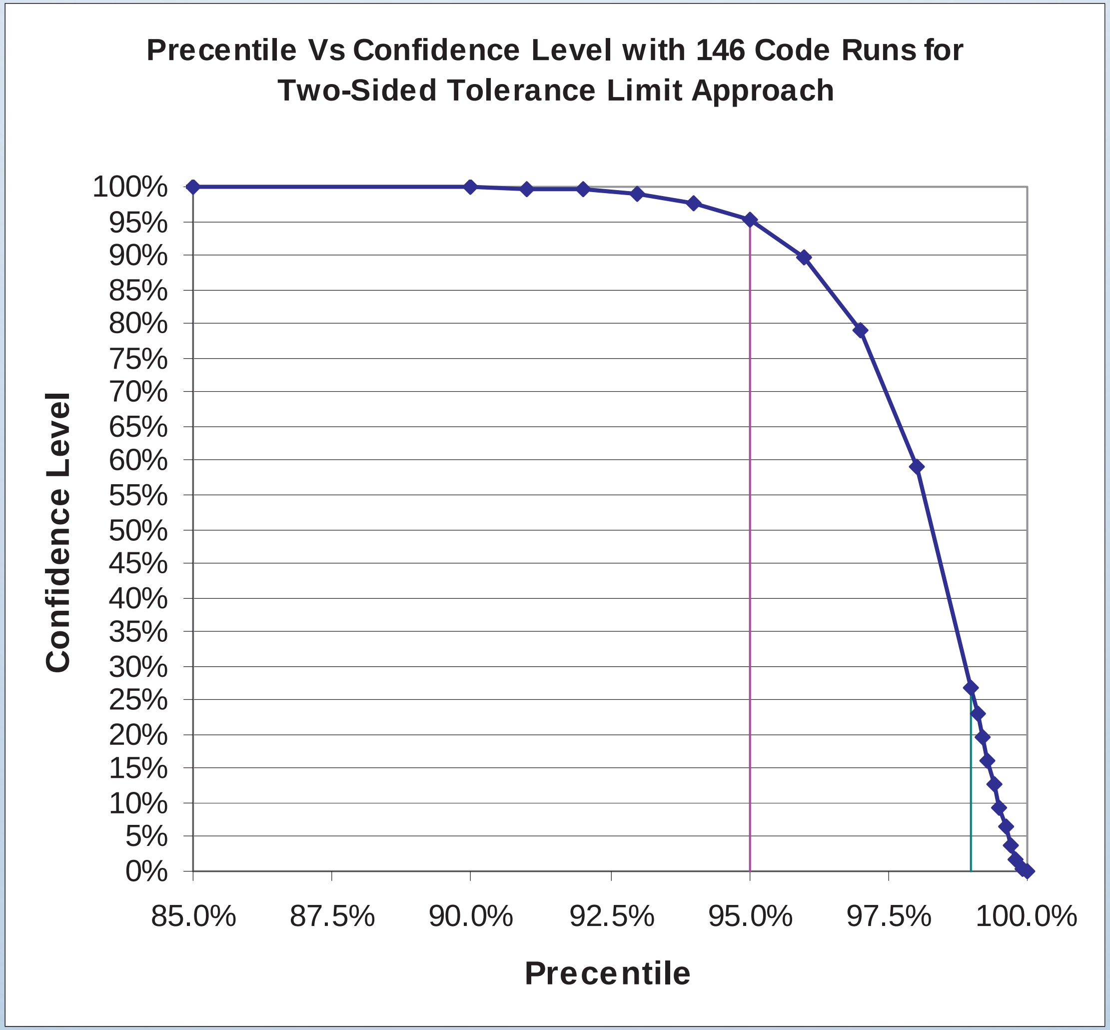
Minimum Number of Code Runs for One-sided 1st order Approach

One-sided 1st order		Percentile in Percentage				
		95.0	96.0	97.0	98.0	99.0
Confidence Level in Percentage	95.0	59	74	99	149	299
	96.0	63	79	106	160	321
	97.0	69	86	116	174	349
	98.0	77	96	129	194	390
	99.0	90	113	152	228	459
	99.5	104	130	174	263	528
	99.9	135	170	227	342	688

Contour Distribution between Confidence Level and Percentile for Given Number of Code Runs (one-sided 1st order)



Relationship between Percentile and Confidence



Summary of Suggested Minimum Numbers of Code Runs at Different Confidence Levels

		Number of Runs for 95th Percentile				
		1sided 1st order	1sided 2nd order	1sided 3rd order	2sided 1st order	2sided 2nd order
Confidence Level (%)	**95.0**	**59**	**93**	**124**	**146**	**221**
	96.0	63	99	130	155	231
	97.0	69	105	138	166	244
	98.0	77	115	148	182	263
	99.0	90	130	165	210	294
	99.5	104	146	182	237	325
	99.9	**135**	**181**	**220**	**301**	**396**

- **Numerical Validation Test Scheme for the Formula Set:**

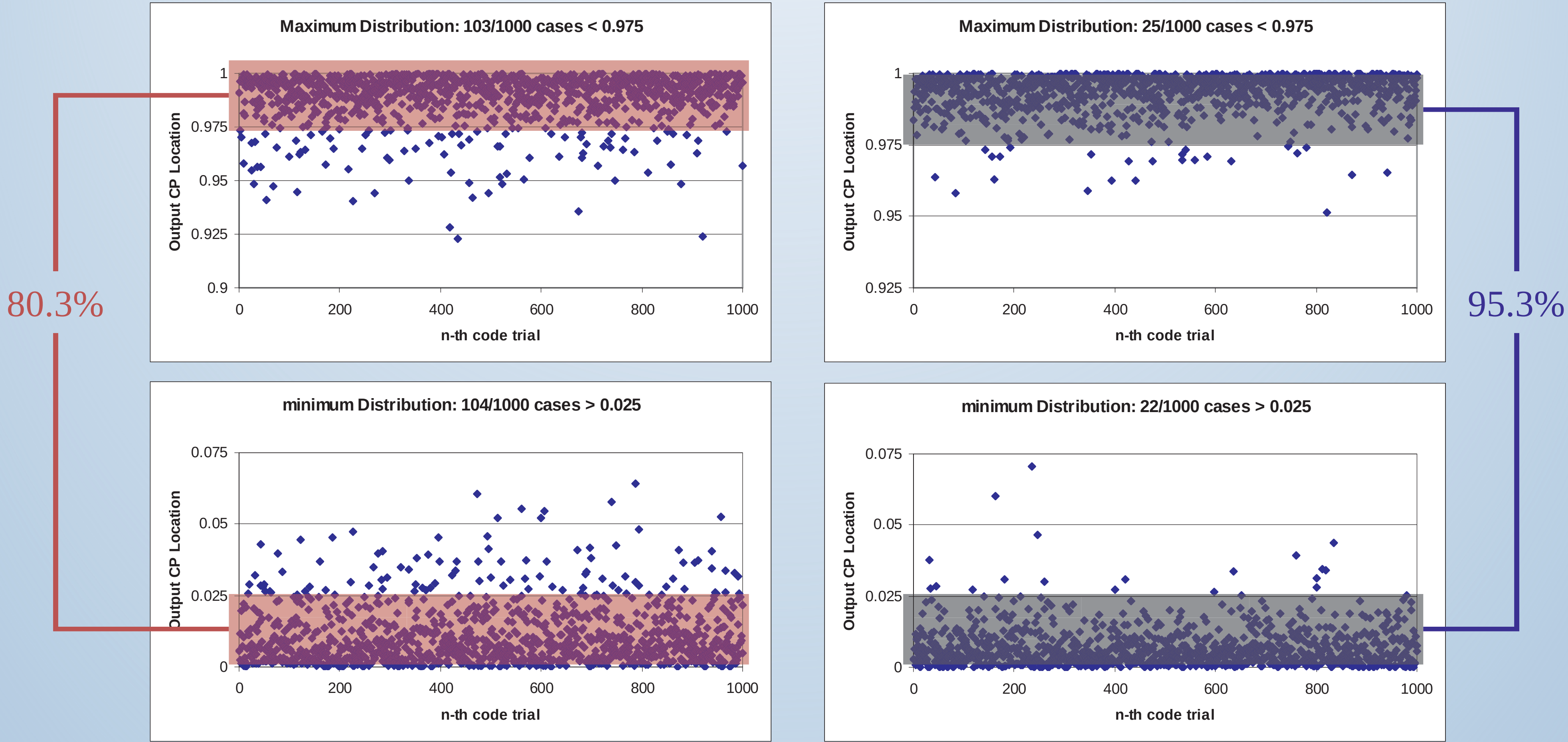
  – An unknown code output parameter is assumed to follow the uniform random distribution (between 0.0 ~ 1.0) as a trial distribution.

  * Note: The Wilks' approach corresponds to distribution free approach.

  – A set of 221 code runs (in case of the two-sided 2^{nd} order) is assumed to constitute the trial code output distribution, and the corresponding 221 code output values are generated using a uniform pseudo-random generator.

  – Then, multiple (for example, 100 or a million) sets of 221 code runs are simulated to investigate the statistical behavior, specifically for the 1^{st} and/or 2^{nd} largest maximum values and the 1^{st} and/or 2^{nd} smallest minimum values.

- **Random Number Generation: 93 vs 146 numbers to simulate code outputs**
- **Test Set: 1000 sets, each contains 93 vs 146 random numbers**



- **Confidence level to satisfy Pr(max>0.975 and min<0.025):**
  - **93 code runs: experimental 80.3%, analytic 81.9%**
  - **146 code runs: experimental 95.3%, analytic 95.1%**

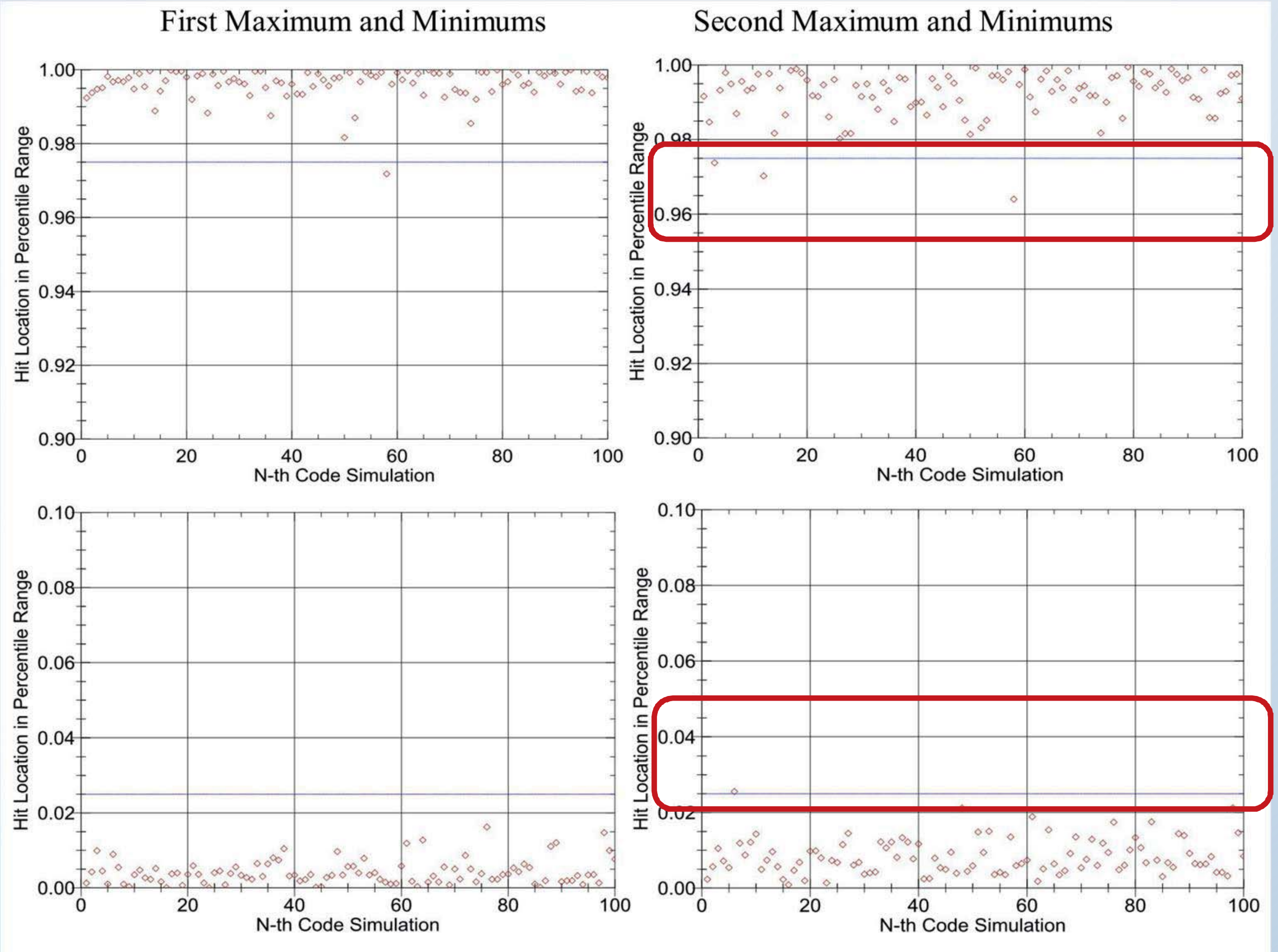
Appropriate Meanings of the Present Wilks' Formula Set

$$one - sided\ 1^{st}\ \ order\ : 1 - \alpha^{n} \geq \beta$$

- the probability that at least one output is outside of the α regardless of its location with a confidence level of β. (General Definition.)
- the probability that the maximum value (or the n^{th} order statistic) will locate on the bigger side than the α, when the α is aligned to the left side, with a confidence level of β.
- the probability that the minimum value (or the 1st order statistic) will locate on the smaller side than the α, when the α is aligned to the right side, with a confidence level of β.

$$two - sided\ 1^{st}\ \ order\ : 1 - \alpha^{n} - n(1-\alpha)\alpha^{n-1} \geq \beta \quad (Suggestion : it\ is\ only\ for\ one\text{ - }sided\ 2^{nd}\ \ order)$$

- the probability that at least two outputs are outside of the α regardless of its location with a confidence level of β. (General Definition.)
- the probability that the 2nd largest value is bigger than the upper tolerance limit of α, or the 2nd smallest value is smaller than the lower tolerance limit α. This definition comes from the 2nd order one-sided approach.
- the probability that the percentile difference between the maximum and minimum to be bigger than the α. In this case, the application of this formula to estimate the locations of the minimum and maximum is invalid since they can be anywhere outside of the α.



* Note: The Percentile in Y axis in the above snapshots correspond to cumulative probability

Numerical Simulation of Max and Min for 2nd Order Two-sided Statistics from 100 Test Cases of Each of 221 Code Run

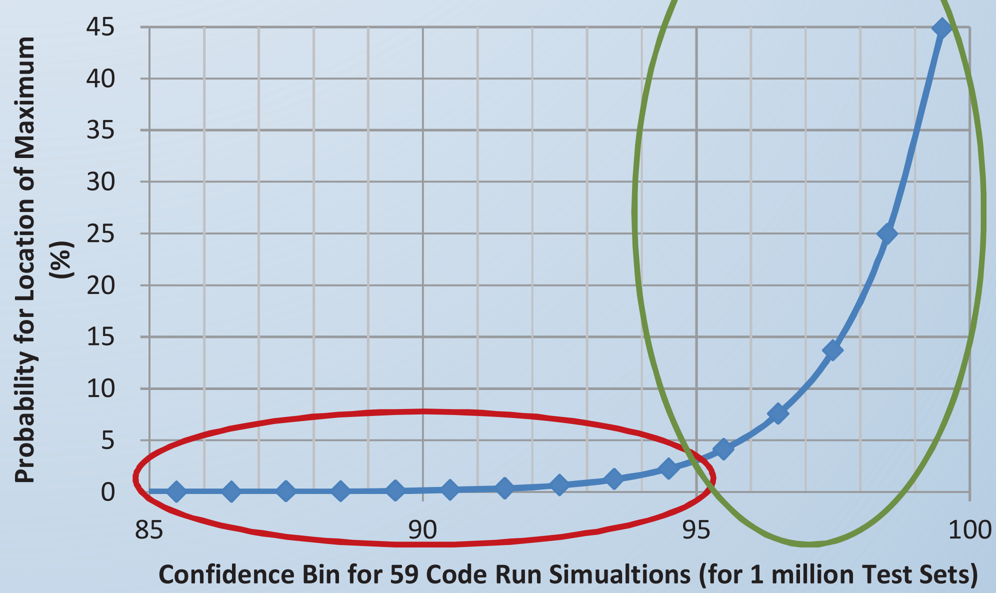
Comparison of Confidence Levels between Numerical Experiment and Theory at the 95th Percentile

Percentile	Approach	N[1)]	Num.Exp.[2)]	Theoretical	Diff (%)
95th	1-sided 1st order	59	95.1622	95.1505	0.01
	1-sided 2nd order	93	95.0305	95.0024	0.03
	2-sided 1st order	146	95.1029	95.0934	0.01
	2-sided 2nd order	221	95.0894	95.1012	-0.01

* Note: 1) N: number of code runs for each set,
2) 1,000,000 tests were performed for each set of N code runs.

Confidence Bin (%)			Frequency out of 1,000,000 Test Sets	Probability for Location of Maximum (%)	Probability for Location of Maximum (%)
0	~	85	99	0.01	4.84
85	~	86	70	0.01	
86	~	87	115	0.01	
87	~	88	268	0.03	
88	~	89	457	0.05	
89	~	90	967	0.10	
90	~	91	1,869	0.19	
91	~	92	3,455	0.35	
92	~	93	6,386	0.64	
93	~	94	12,098	1.21	
94	~	95	22,620	2.26	
95	~	96	41,169	4.12	95.16
96	~	97	75,654	7.57	
97	~	98	136,742	13.67	
98	~	99	249,531	24.95	
99	~	100	448,500	44.85	

One-Sided 1st Order Result for Sets of 59 Code Runs



Discussion

- **Which number is more appropriate to meet the 95th / 95 % requirement?**
  - We are relying on higher confidence level in reality when we are using higher order statistics. (It could be thought otherwise of course.) The 1st order approach may need be more credited than the higher order approach.
  - A set of Wilks' combinatorial formula were newly suggested specifically for two-sided 1st and 2nd order approaches. The approach might need more attention at the application level.

- **Can we all agree on the 95th / 95 % practice from the safety perspective?**
  - Historically, the 95 % probability level, e.g., 95th percentile combined with the confidence level of 95 in the nuclear field seems to stem from the standard engineering practice and be based on the US NRC's acceptance as stated in the regulatory guide 1.157, which guides the best-estimate and uncertainty approach to the ECCS analysis.
  - The present practice of crediting the confidence level of 95% might need some attention in that it means that there exists 1 out-of 20 missing possibility.
  - Maybe we cannot simply say that it is very acceptable to allow the 5 % of maybe-dangerous conclusion.
  - It seems reasonable to take into account the importance of the confidence level more than the percentile.

Wrapup

- The introduction of Wilks' formula by GRS might be one of the most significant contribution.
- Through the review of the BEMUSE project results, our observation is as follows:
  - The tolerance limit evaluation approach would be applicable to not only the safety analysis discipline but wider range of disciplines in the near future.
  - A more in-depth understanding of the tolerance limit approach might be necessary at the working group levels.
  - We suggested an improved formula set to determine the size of statistically meaningful minimum code simulations.
  - The present practice of using a more than enough number of theoretically derived minimum numbers of code runs is concluded in two ways;
    a) it is reasonable in a sense that it ensures a higher confidence level than the 95 % in terms of the 1st order, however,
    b) it is unreasonable because it may not produce the intended results.
  - From the safety perspective, the present practice of crediting the 95th percentile looks reasonable but the 95 % confidence level may or may not be high enough to ensure nuclear safety analysis results.

- **One final suggestion is that there should be a consensus for proper applications of the tolerance limit evaluation approach between different organizations, disciplines.**

THANK YOU.

An Integrated Approach for Characterization of Uncertainty in Complex Best Estimate Safety Assessment

Presented
By

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BCN Workshop, 16 Nov. 2011

Acknowledgments

- Co-authors: M. Pourgolmohamad, Ph.D, PE, Currently Currently An Assistant Professor of Mechanical Engineering with Sahand University of Technology, Tabriz, Iran.
- Ali Mosleh, Professor & Director of Center for Risk and Reliability, University of Maryland, College Park, MD
- This work was performed under a cooperative research agreement between the Center for Risk and Reliability at the University of Maryland and the US Nuclear Regulatory Commission during 2005-2007.

Major Publications on this Approach

- Integrated Methodology for Thermal-Hydraulic Code Uncertainty Analysis with Application, M. Pourgolmohamad, M. Modarres, A. Mosleh, Nuclear Technology, Volume 165, Number 3 · March 2009 · Pages 333-359
- Methodology for the Use of Experimental Data to Enhance Model Uncertainty Assessment in Thermal Hydraulics Codes, M. Pourgolmohamad, A. Mosleh, M. Modarres, Reliability Engineering and System Safety, Reliability Engineering and System Safety 95 (2010) 77–86.
- Structured Treatment of Model Uncertainty in Complex Thermal-Hydraulics Codes; Technical Challenges, Prospective and Characterization, M. Pourgolmohamad, Ali Mosleh, M. Modarres, Nuclear Engineering and Design, Volume 241, Issue 1, January 2011, Pages 285-295.
- 10 other conference or workshop papers

Motivation

- We are a PSA research group interested in assessment of risks and use of risk information in safety regulations
- TH and other mechanistic codes are used in many PSA studies (success criteria for safety systems such as ECCS, PTS studies, Fire Risks, etc.)
- USNRC revised ECCS licensing rules to allow the use of best estimate computer code plus uncertainty
- Assessment of uncertainties in PSAs are critical
- The approach has been developed in the context of applications in risk-informed and other PSA needs and applications

Outline

- Scope of Research
- Overview on IMTHUA methodology
- Complexity and Structure of TH Codes
- Multi-Model Uncertainty Analysis
  - ✓Single Model
  - ✓Alternative Models
- Application of the Methodology to LOFT LBLOCA

Steps Involved

- ✓ Input Phase
  - Modified PIRT
  - Code Models and Parameters
  - Inputs and Model Structure Uncertainty Quantification
- ✓ Alternative Models
  - Dynamic Model Switching
  - Model Mixing
- ✓ Output-Based Bayesian Updating
  - Approach
  - Data Availability and Treatment
    - Model Uncertainty
    - Partially Relevant Data

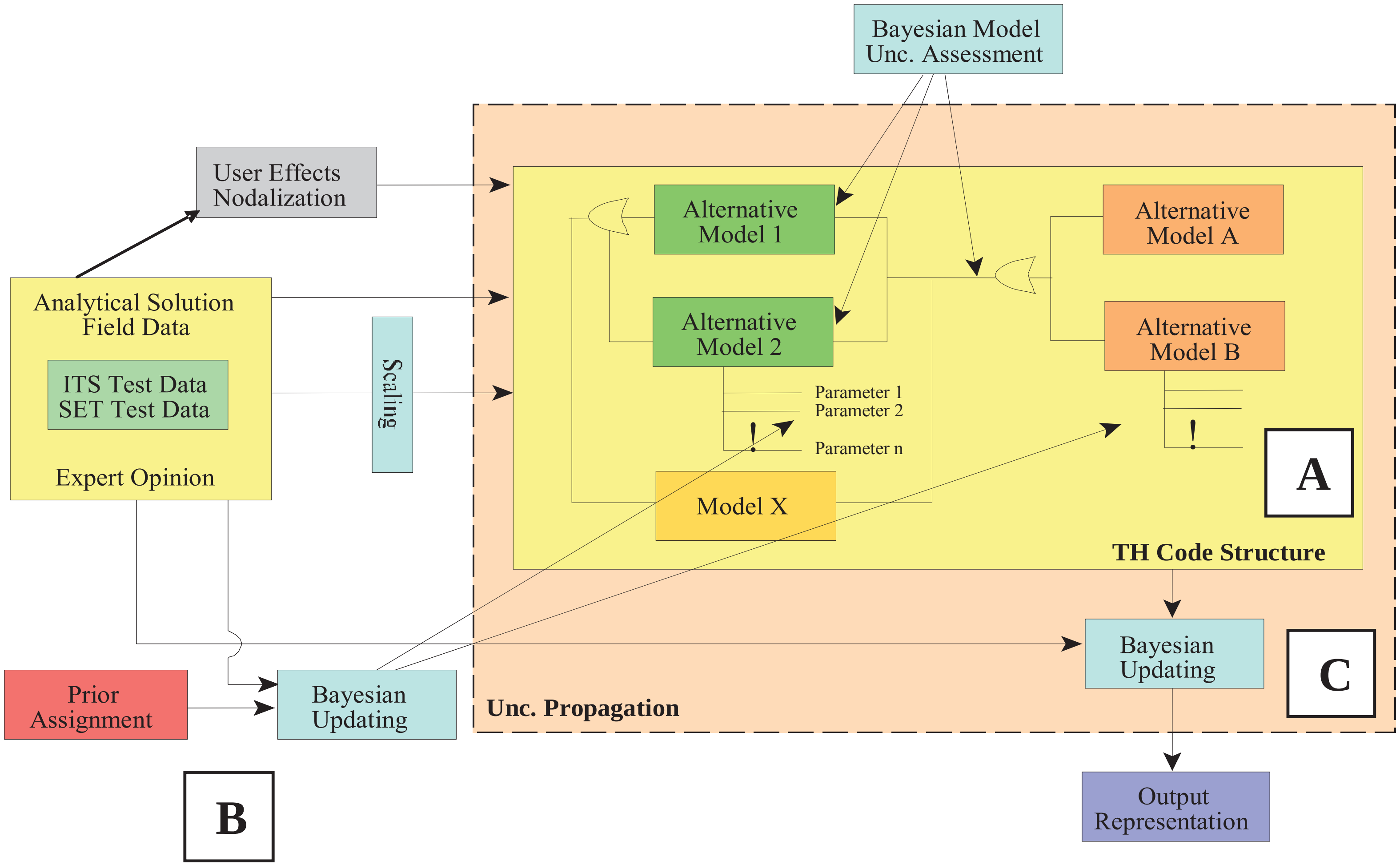
Scope

- Integrated Methodology for TH Uncertainty Analysis (IMTHUA) : An Amalgam of Promising Features from Existing Methodologies
- Use of Most Available Information to Assess Uncertainties Related to
  - ✓ Boundary/Initial Conditions
  - ✓ Models, Sub-Models and Corresponding Parameters
  - ✓ Output Updating Using Bayesian Inference

Aspects of TH Codes Affecting Uncertainties

- ✓ Limited user control over code structure
- ✓ Limited and/or partially relevant data / information about models, sub-models, and correlations, such as HTC
- ✓ Large number of interacting models and correlations (thousands)
- ✓ Only a small portion of the code models may be active during each time step, depending on the underlying simulation and system conditions
- ✓ Many horizontal and vertical flow regime phases in the code calculation, with fuzzy borders between them
- ✓ Inability to precisely solve field equations for specific configurations due to coarse average nodes
  - For example, choked flow model is called in TH codes calculation when the results of momentum equation calculation is unsatisfactory. The code calls for a choked flow model for velocity calculation and replaces it with the previous calculation. For better resolution, TH codes are recently coupled with CFD codes for more accurate calculations where needed.

Overall Methodology Overview



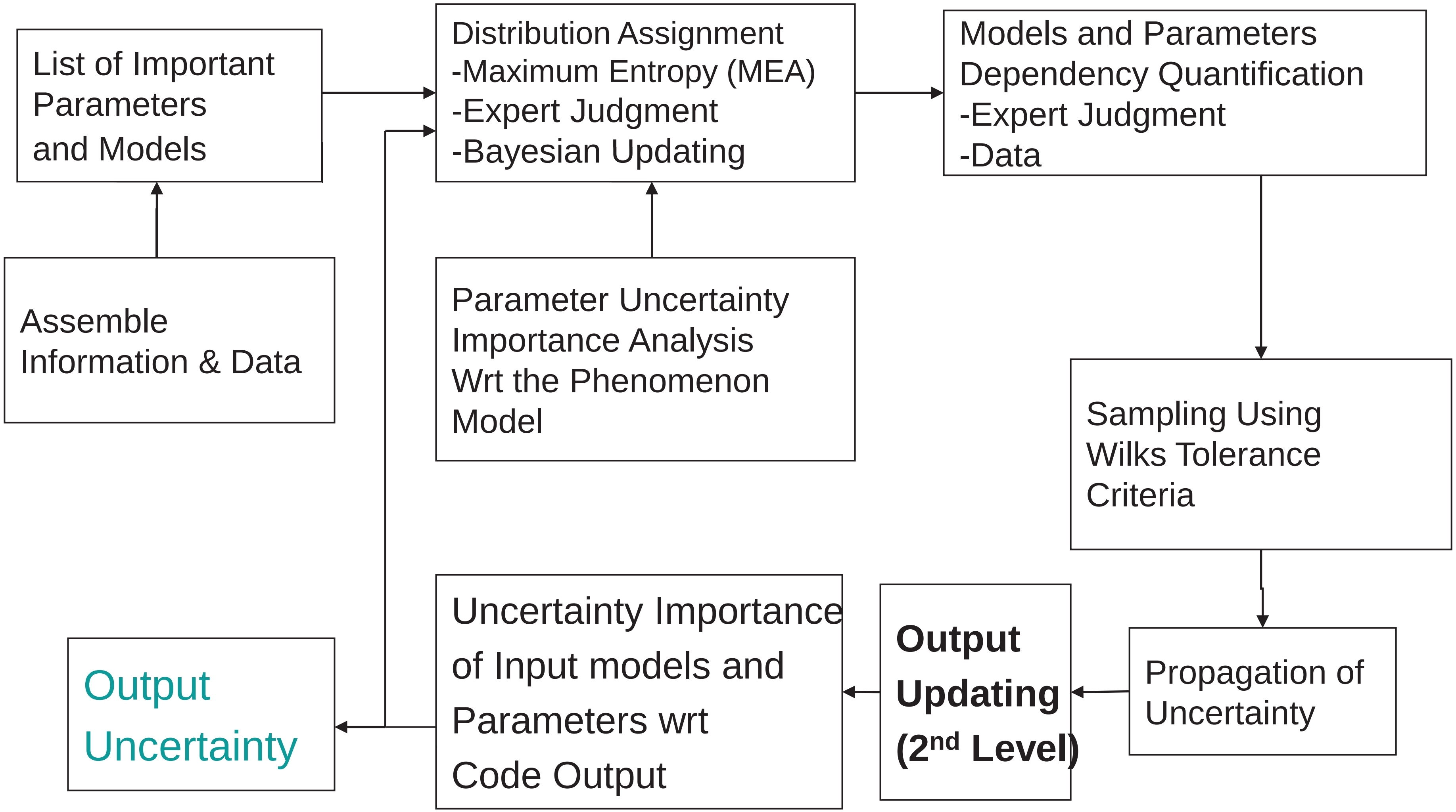
Overall Methodology Overview (Cont.)

- Treatment of the code structure uncertainty (the White-Box Approach): Step A. Key objective: Explicit quantification of uncertainties due to model form (structure) as well as model parameters.
- Applied both at the sub-model levels and also the entire TH code (Step C).
- Input parameter uncertainty quantification is performed via the Maximum Entropy and/or and expert judgment methods, depending on the availability and type of information (Step B).
- Hybrid of input-based and output-based uncertainty Assessment (Step C) uncertainty analysis: Therefore IMTHUA is a two-step uncertainty quantification.

Overall Methodology Overview (Cont.)

- Modified PIRT: This is a two-step method that identifies and ranks phenomena based on their: (a) TH influence (using AHP), and (b) Uncertainty ranking based on an expert judgment procedure. See: Pourgolmohamad M, Modarres M., Mosleh A. Modified Phenomena Identification and Ranking Table (PIRT) For Uncertainty Analysis, Proceedings of 14th International Conference on Nuclear Engineering, July 17-20, 2005, Miami, Florida, USA.
- Uncertainty propagation through the use of Wilks' tolerance limits sampling criteria to reduce the number of Monte Carlo iterations for the desired accuracy.

Assessment & Propagation of Uncertainties in Models & Parameters



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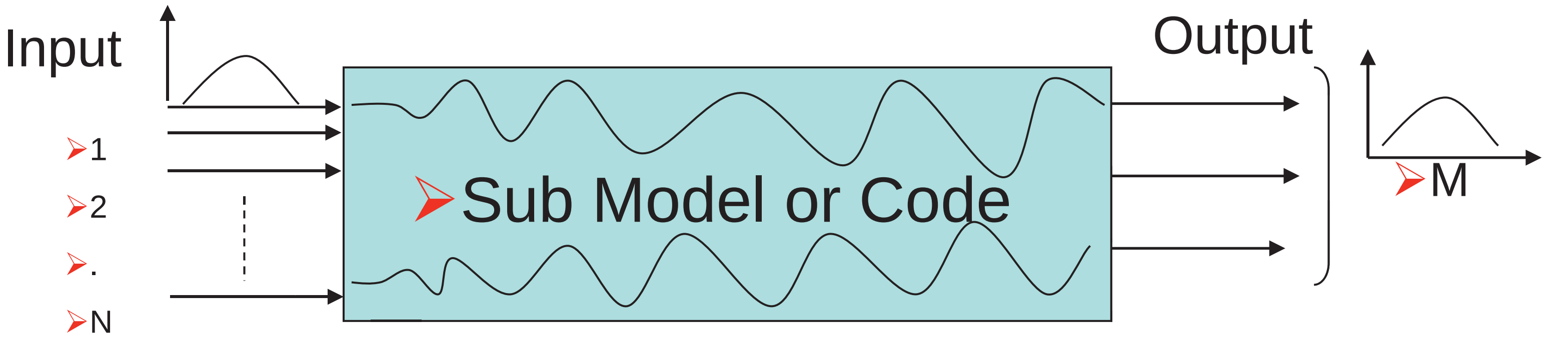
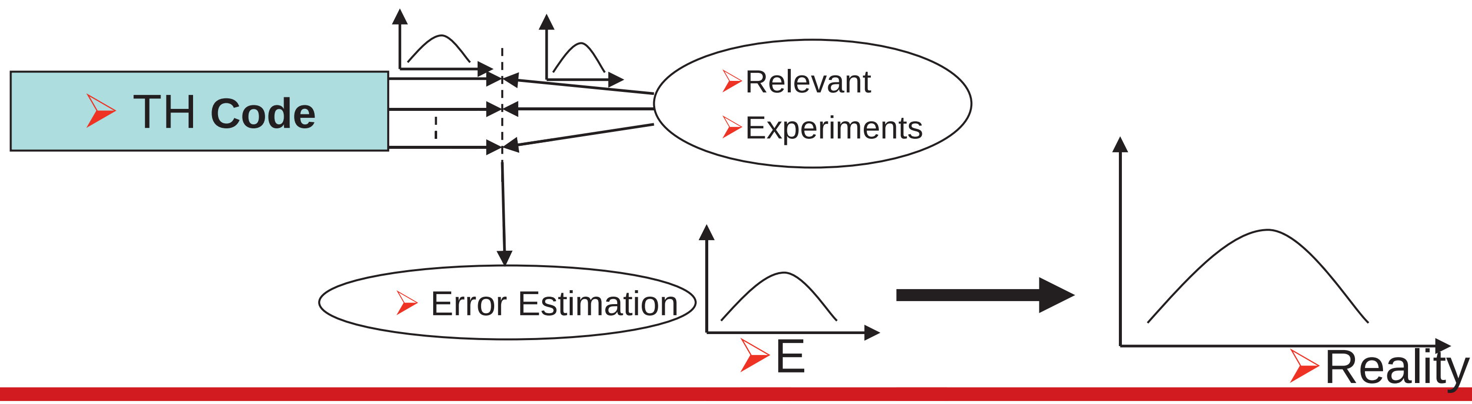
Model Output and Error Uncertainties

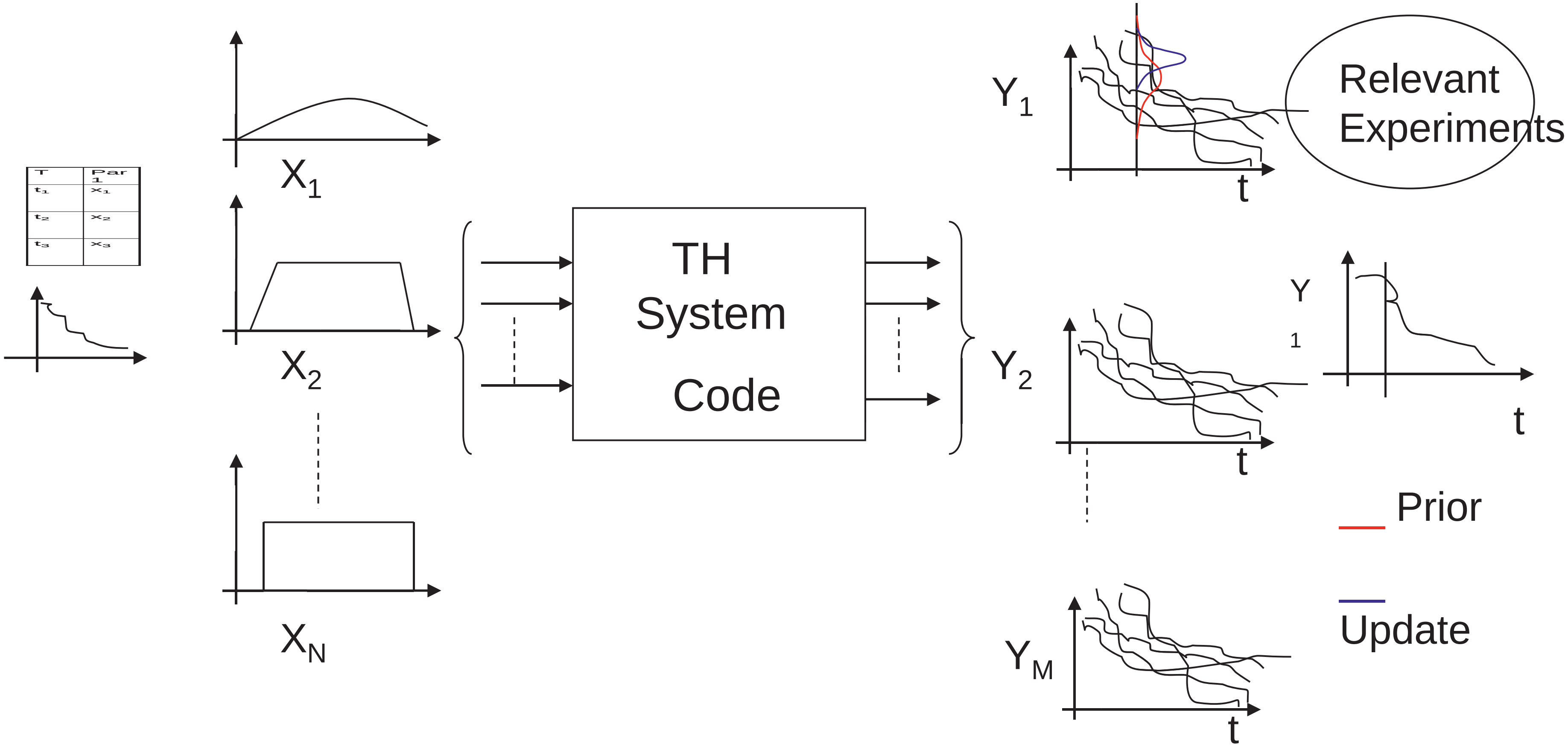
- Model output uncertainty



- Model Error Uncertainty



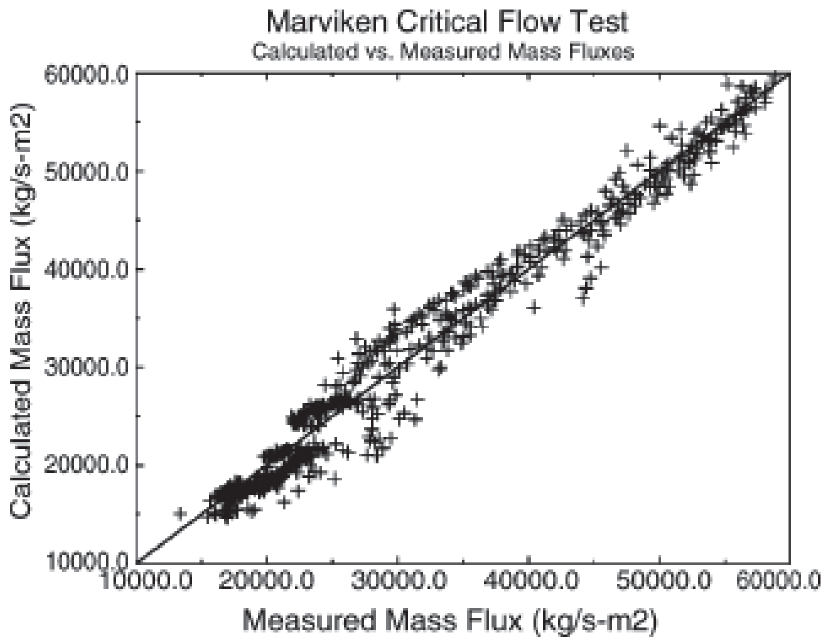
Summary of The Methodology



Singe Model Uncertainty Treatment

- Multiplicative Error
- Bias Consideration
- Uncertainty Treatment for Code Structure

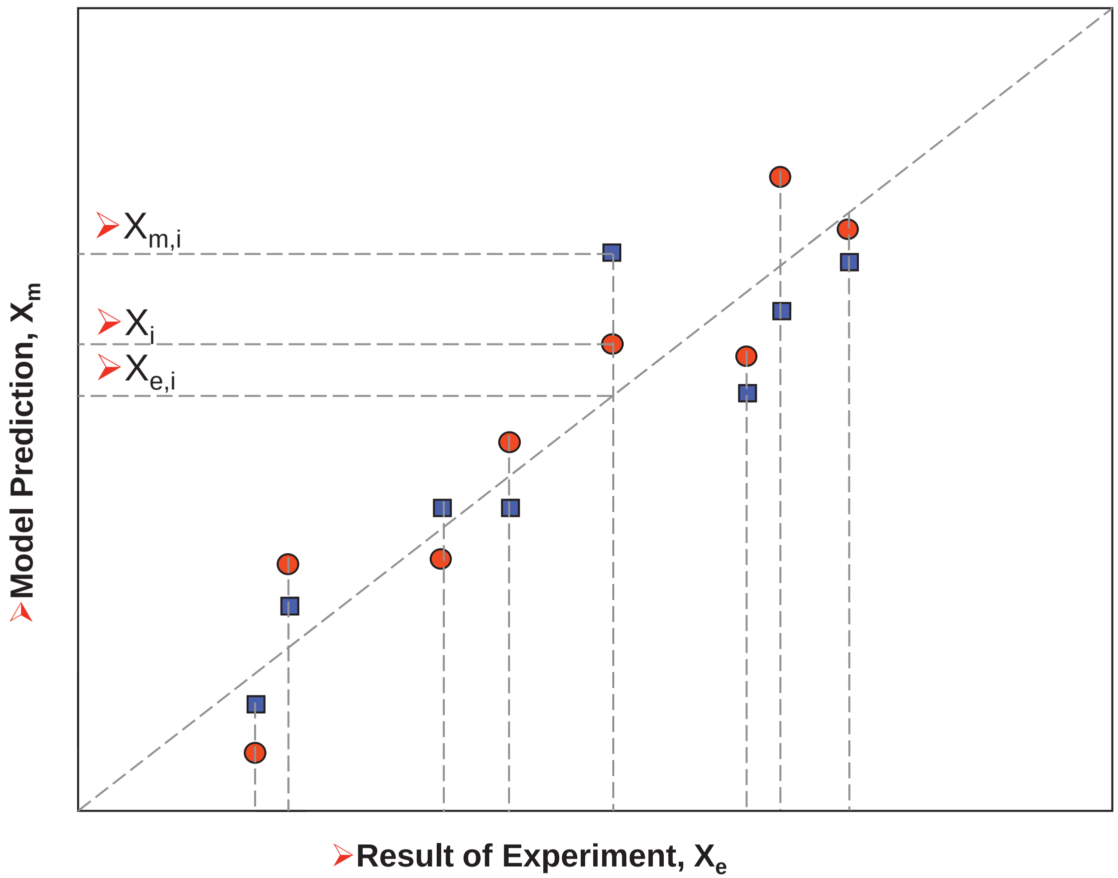
$$R_{in} = \frac{\text{Measured Flow Rate}}{\text{Predicted Flow Rate}}$$



✓ E.g., TRAC natural choking model has an average bias of 1.2

Accounting for Model Uncertainty

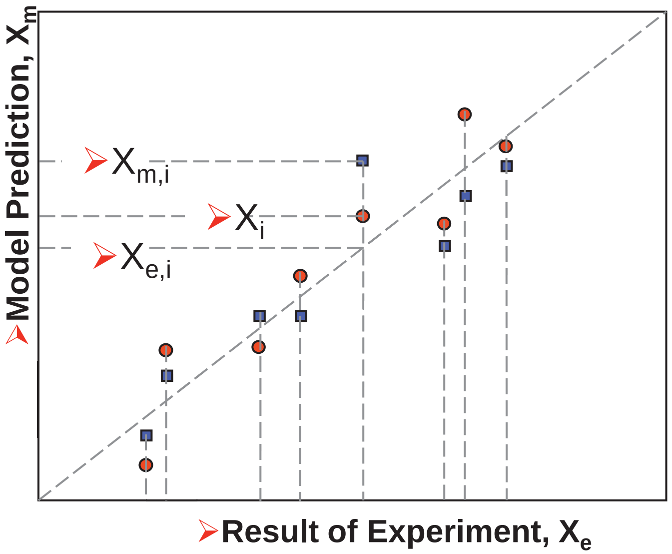
- Scatter of Model Prediction vs. Experimental Measurement



Multiplicative Error: Approach and Assumptions

- The model prediction (output), result of experiment and real value of interest have the same sign (all positive or all negative)
- The ratio of real value and experimental results (or data) is a random variable with lognormal distribution for which the confidence bounds are known (Experimental Accuracy)
- The ratio of real value and model prediction (output) is a random variable with lognormal distribution with parameters to be determined
- The ratio of model predictions and results of experiment is a function of the two random variables introduced earlier. The distribution of this random variable is lognormal and will be used to represent the likelihood of data
- The distribution of real quantity of interest given a model prediction will be a lognormal distribution

Multiplicative Error Model



$$\frac{X_i}{X_{e,i}} = F_{e,i} \quad ; \quad F_e \sim LN(b_e, \sigma_e) \qquad (1)$$

$$\frac{X_i}{X_{m,i}} = F_{m,i} \quad ; \quad F_m \sim LN(b_m, \sigma_m) \qquad (2)$$

where :

X : Real Quantity

X_e : Result of experiment

X_m : Model prediction

F_e : The error factor for experimental data

F_m : The error factor for model predictions

b_e, σ_e : Mean and SD of experimental error factor

b_m, σ_m : Mean and SD of model error factor

Substituting (1) in (2) :

$$\left.\begin{array}{l} F_{e,i} X_{e,i} = F_{m,i} X_{m,i} \\ \dfrac{X_{e,i}}{X_{m,i}} = \dfrac{F_{m,i}}{F_{e,i}} = F_{t,i} \\ \text{Independency of } F_m, F_e \end{array}\right\} \Rightarrow F_t \sim LN\left(b_m - b_e, \sqrt{\sigma_m^2 + \sigma_e^2}\right)$$

Multiplicative Error: Bayesian Posterior

- N = M

$$f(b_m, \sigma_m | X_{e,i}, X_{m,i}, b_e, \sigma_e) = \frac{f_0(b_m, \sigma_m) \times \left[L(X_{e,i}, X_{m,i}, b_e, \sigma_e | b_m, \sigma_m)\right]^{\beta}}{\int_{\sigma_m}\int_{b_m} f_0(b_m, \sigma_m) \times \left[L(X_{e,i}, X_{m,i}, b_e, \sigma_e | b_m, \sigma_m)\right]^{\beta} db_m d\sigma_m}$$

where:

$$L(X_{e,i}, X_{m,i}, b_e, \sigma_e | b_m, \sigma_m) = \prod_{i=1}^{n} \frac{1}{\sqrt{2\pi}\left(\frac{X_{e,i}}{X_{m,i}}\right)\sqrt{\sigma_m^2 + \sigma_e^2}} e^{-\frac{1}{2}\times\frac{\left[\ln\left(\frac{X_{e,i}}{X_{m,i}}\right) - (b_m - b_e)\right]^2}{\sigma_m^2 + \sigma_e^2}}$$

$f_0(b_m, \sigma_m)$: Prior Joint Distirbution of Parameters

$f(b_m, \sigma_m | X_{e,i}, X_{m,i}, b_e, \sigma_e)$: Posterior Joint Distirbution of Parameters

$\beta = 0$ *to* 1 *with* 0 *for least relevant and* 1 *for fully relevant*

Given a model prediction such as X_m the distribution of the real value X will be:

$$\left.\begin{array}{l} X_m \text{ given as model prediction} \\ F_m \sim LN(b_m, \sigma_m) \\ X = F_m X_m \end{array}\right\} \Rightarrow X \sim LN\left(\ln(X_m) + b_m, \sigma_m\right)$$

Multiplicative Error: Bayesian Posterior (Cont.)

✓ N ≠ M

$$F_t \sim \int_{b_m, s_m} LN\left(b_m - b_e, \sqrt{\sigma_m^2 + \sigma_e^2}\right) \cdot g\left(b_m, s_m\right) db_m ds_m$$

$$f\left(b_m, s_m \mid X_{e,k}, X_{m,i}, b_e, s_e\right) =$$

$$\int_\omega \bar{f}\left(b_m, s_m \mid \omega\right) \frac{\prod_{k=1}^{N}\left(\prod_{i=1}^{M_k} \int_\theta L\left(X_{e,k}, X_{m,i}, b_e, s_e \mid b_m, s_m\right)\ f\left(b_m, s_m \mid \omega\right) d\theta\right)\ \pi_o\left(\omega\right)}{\int_\omega \prod_{k=1}^{N}\left(\prod_{i=1}^{M_k} \int_\theta L\left(X_{e,k}, X_{m,i}, b_e, s_e \mid b_m, s_m\right)\ f\left(b_m, s_m \mid \omega\right) d\theta\right)\ \pi_o\left(\omega\right)\ d\omega} d\omega$$

$where:$

$$L(X_{e,i}, X_{m,i}, b_e, \sigma_e \mid b_m, \sigma_m) = \prod_{i=1}^{n} \frac{1}{\sqrt{2\pi}\left(\dfrac{X_{e,i}}{X_{m,i}}\right)\sqrt{\sigma_m^2 + \sigma_e^2}} e^{-\frac{1}{2} \times \frac{\left[\ln\left(\frac{X_{e,i}}{X_{m,i}}\right) - (b_m - b_e)\right]^2}{\sigma_m^2 + \sigma_e^2}}$$

$f_0(b_m, \sigma_m)$: Prior Joint Distribution of Parameters

$f(b_m, \sigma_m \mid X_{e,i}, X_{m,i}, b_e, \sigma_e)$: Posterior Joint Distribution of Parameters

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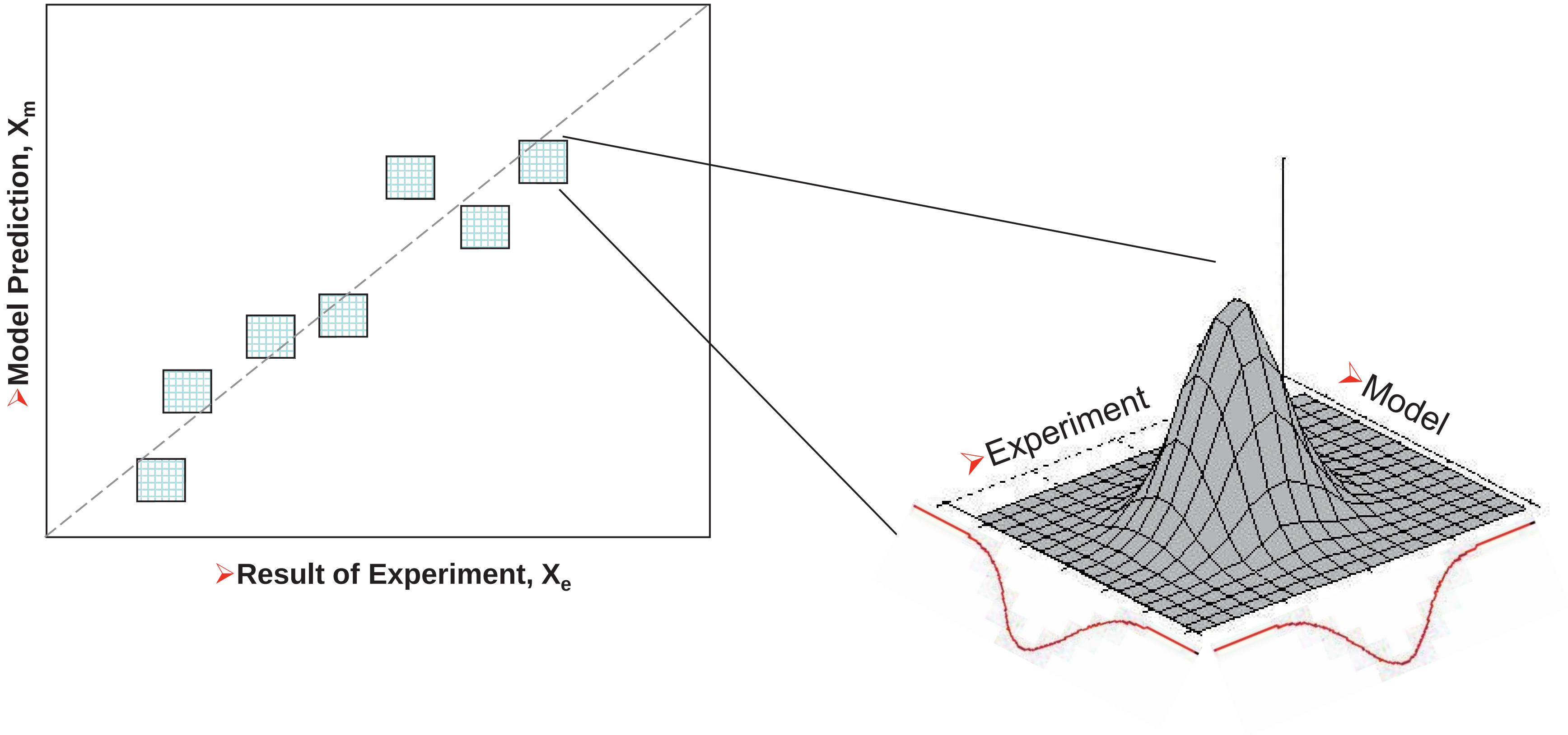
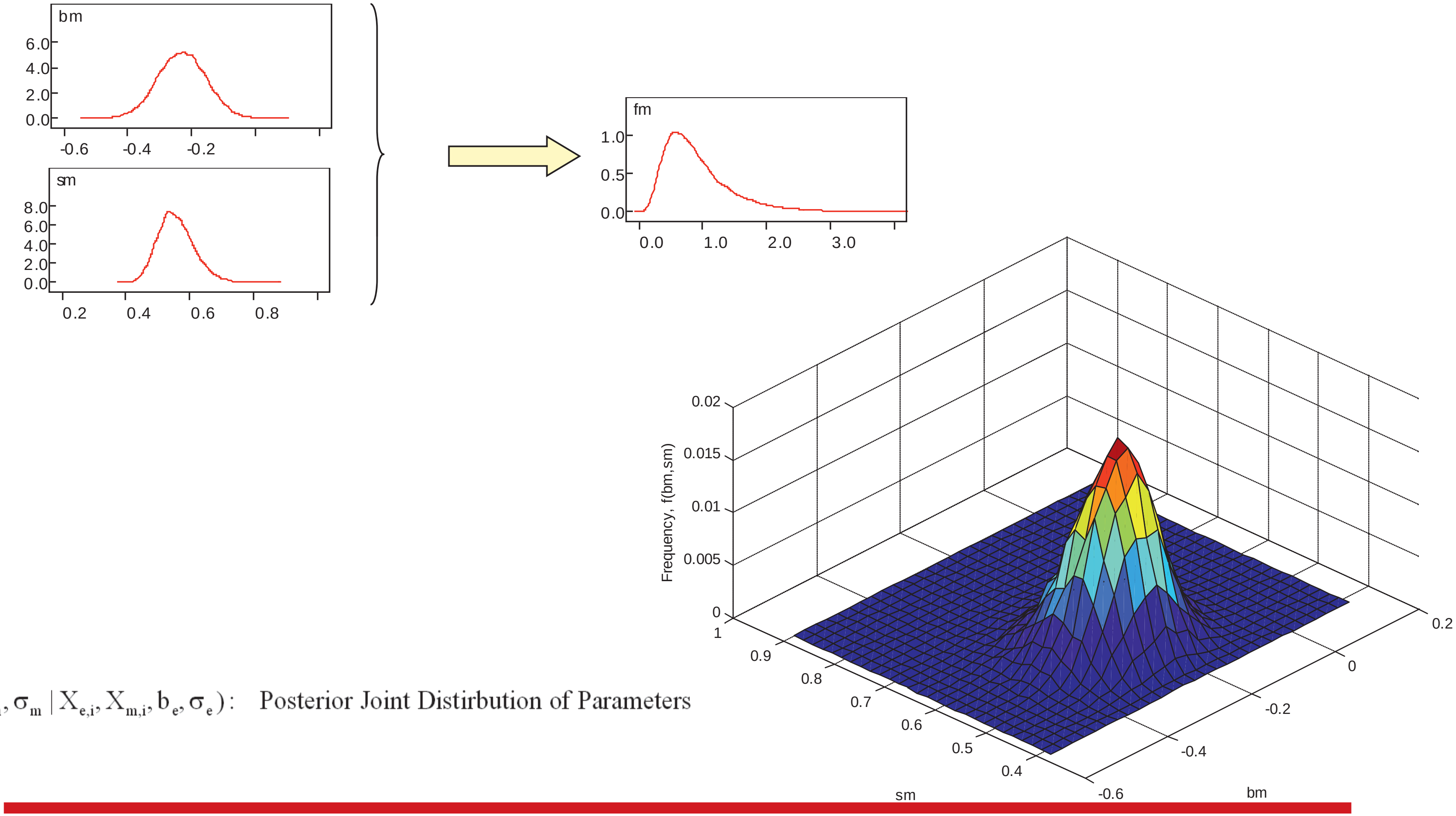
Including Model Uncertainty

➢When Both Model Output and Experimental Data Are Uncertain:



Heat Flux Model Updating Using WinBUGS

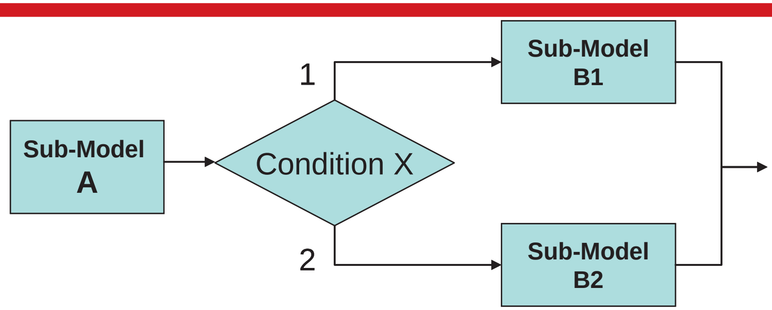


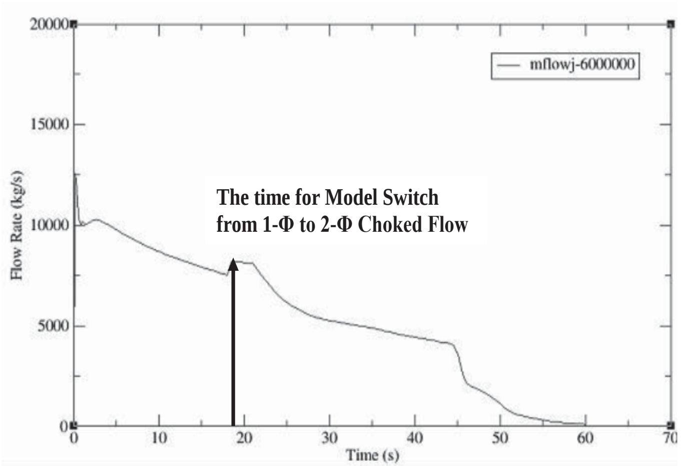
$f(b_m, \sigma_m \mid X_{e,i}, X_{m,i}, b_e, \sigma_e)$: Posterior Joint Distirbution of Parameters

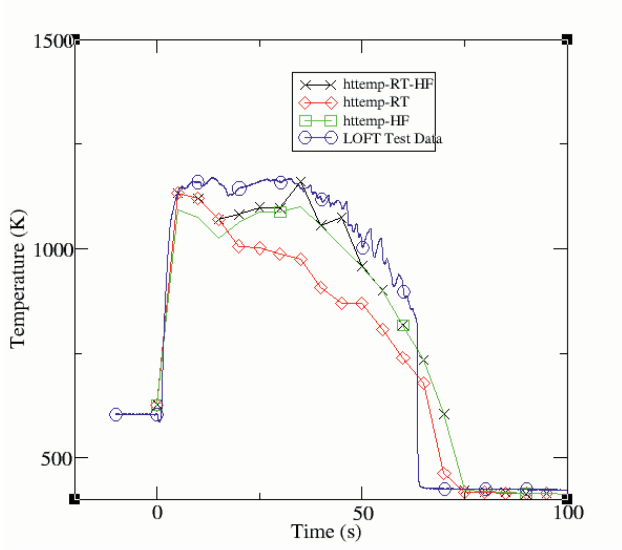
Alternative Models Treatments

- Dynamic Model Switching (Treatment of Switching Time/Condition Uncertainty)
- Recommended Code Option
- Model Mixing (Treatment by Weighted Probability)

Dynamic Model Switching

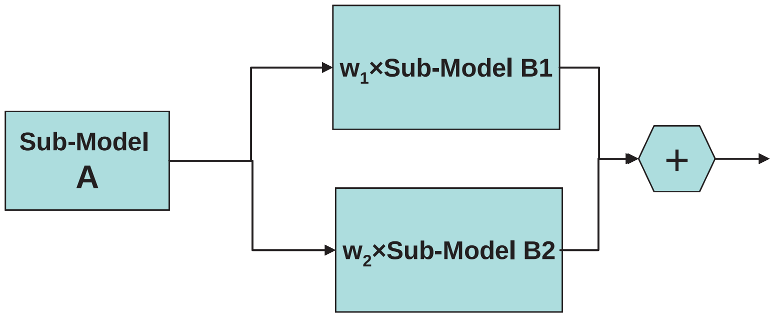


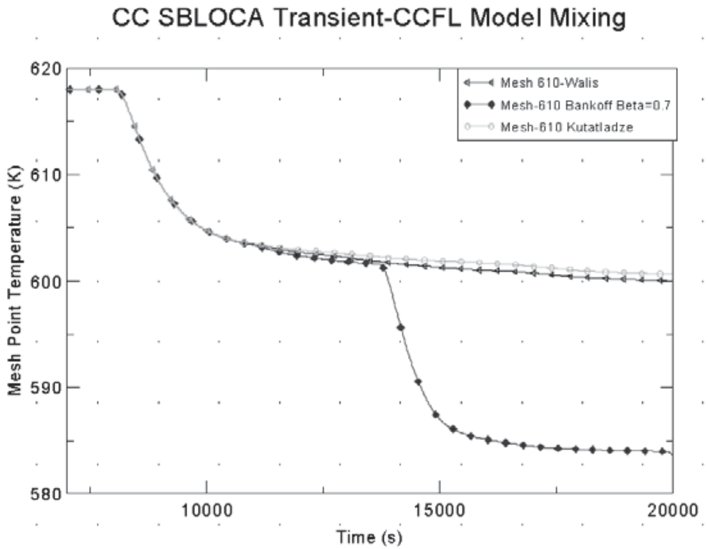




- Model Switch from 1-Φ Choked Flow to 2- Φ Choked Flow-Marviken Blowdown
- Model Switch by Code or User for Henry-Fauske and Henry-Trap Choked Flow Model

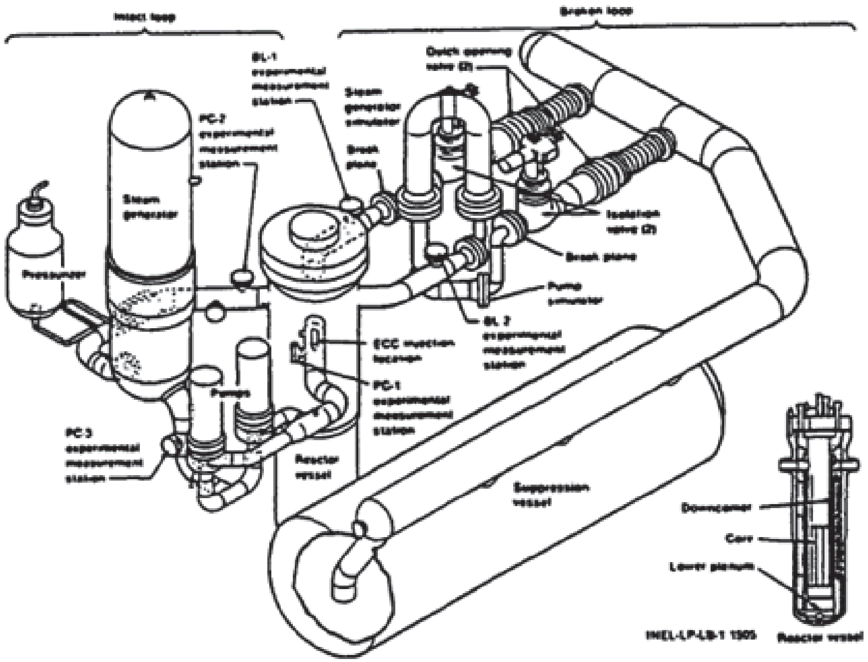
Model Mixing





➢ Inference requires careful assessment

LOFT Application Test LB-1 Facility



Item	LOFT
Fuel rod number	1300
Length (m)	1.68
Inlet flow area (m3)	0.16
Coolant volume (m3)	0.295
laximum linear heat generation rate (kW/m	39.4
Coolant temperature rise (K)	32.2
Power (MW)	36.7
Peaking factor	2.34
Power/coolant volume (MW/m3)	124.4
Core volume/system volume	0.038
Mass flux (Kg/s-m2)	1248.8
Core mass flow/system volume (Kg/s-m3)	25.6

Initial Conditions and Scenario Sequence of Time

LOFT measured initial conditions LB-1	
Parameter	***LB-1***
Reactor Power (MW)	***49.3***
Low Pressure Scram Set Point (MPa)	***14.5***
Intact-loop Mass Flow(kg/s-m2)	***305.8***
Hot-leg Pressure (Mpa)	***14.77***
Hot-leg Temperature (°K)	***586.1***
Cold-leg Temperature (°K)	***556.6***
Pump Speed (rad/s)	***209***
Pressurizer Steam Volume (m^3)	***0.38***
Pressurizer Liquid Volume (m)	***0.55***
Steam-generator Pressure (MPa)	***5.53***
Steam-generator Mass Flow(kg/s)	***25.4***
Accumulator Pressure (MPa)	***4.21***
Accumulator Temperature (°K)	***305***
Accumulator Initial Level (m)	***2.31***
Accumulator Level at End of Discharge (m)	***1.75***
Accumulator Liquid Level Change (m)	***0.56***
Accumulator Liquid Volume Discharged (m^3)	***0.76***
Accumulator Initial Gas Volume (m^3)	***0.65***
Accumulator Initial Gas/Liquid Fraction	***0.85***

- Scenario Specification
  - ✓ High Power Fuel Assembly
  - ✓ 200% Cold Leg Break Test
  - ✓ Higher Reactor Power (49.3 MW) and Loop Flow
  - ✓ Inactivated High Pressure Injection
  - ✓ Intact Loop Pumps with Fly Wheel Disconnected Fly Wheel at Pump Trip

LOFT Test LB-1 Sequence of Event Timing		
Event	***Measured***	***Code Results***
Break initiated (s)	***0***	***0***
Reactor scrammed (s)	***0.13***	***0.13***
Primary-coolant pumps tripped (s)	***0.63***	***0.63***
Pressurizer emptied (s)	***Instrument failure***	***15.5***
Accumulator A injection initiated (s)	***17.4***	***14***
Reflood Tripped On (s)	***NA***	***0***
HPIS injection initiated (s)	***NA***	***NA***
LPIS injection initiated (s)	***24.8***	***24.8***
Maximum cladding temperature (°K)	***1170***	***1050***

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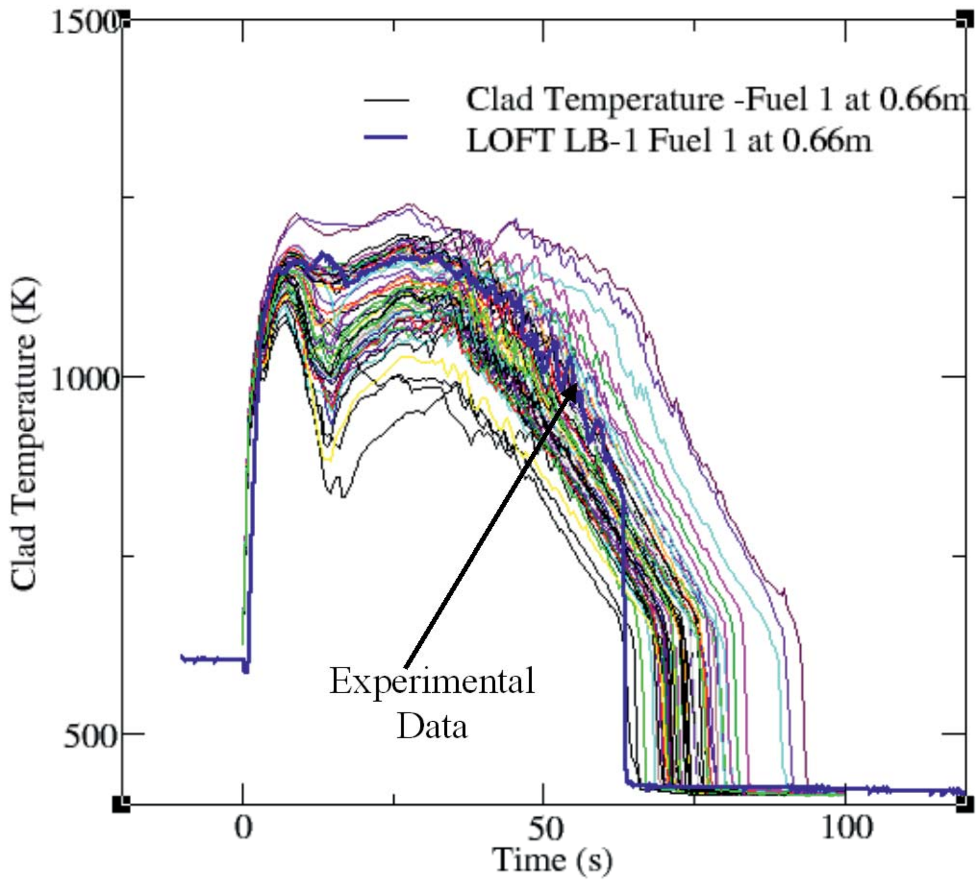
Code Models and Parameters

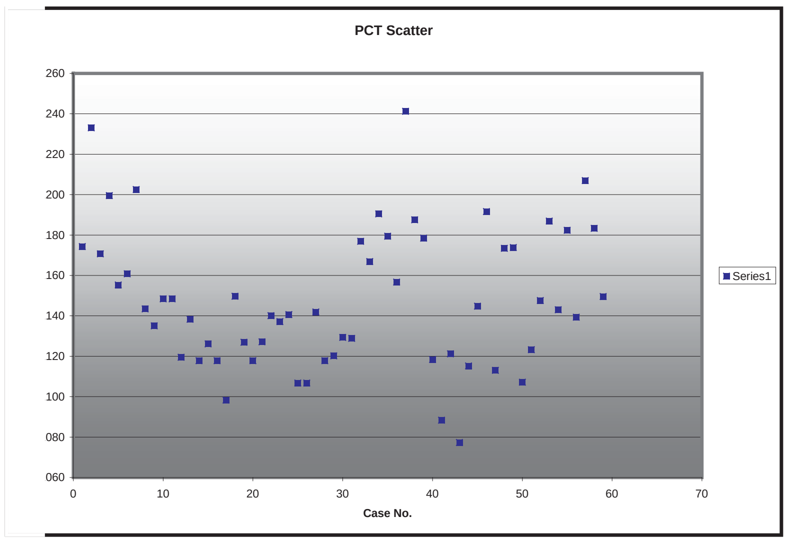
Choked Flow	**2-Phase Model Multiplier** **1-Phase model multiplier**
Post CHF Heat Transfer	**Gap Conductance Model** **-**Fuel Conductance Input Table in Inputdeck
Pressurizer Level	**Level Controller Card in the Inputdeck** -Measurement Error 1.04 +/- 4 cm
Core Power	**Power table** -Measurement error 49.3 Mwt+/-1.3 MW_t **Fuel and Cladding Thermal Conductivity**
Entrainment	**Hydraulics Diameters (Hot Leg, Downcomer, etc)**
Peaking Factor	**Radial**

Sample Distributions

Uncertainty Propagation-Modified PIRT LOFT LBLOCA

LOFT LOB-1 Uncertainty Analysis





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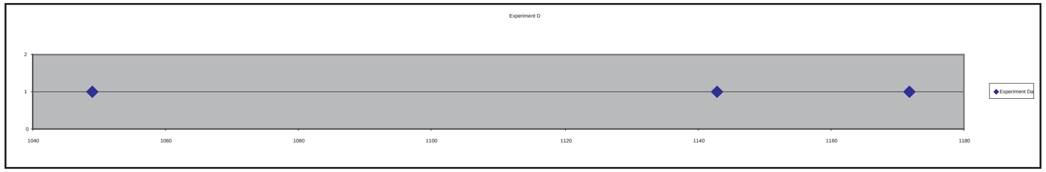
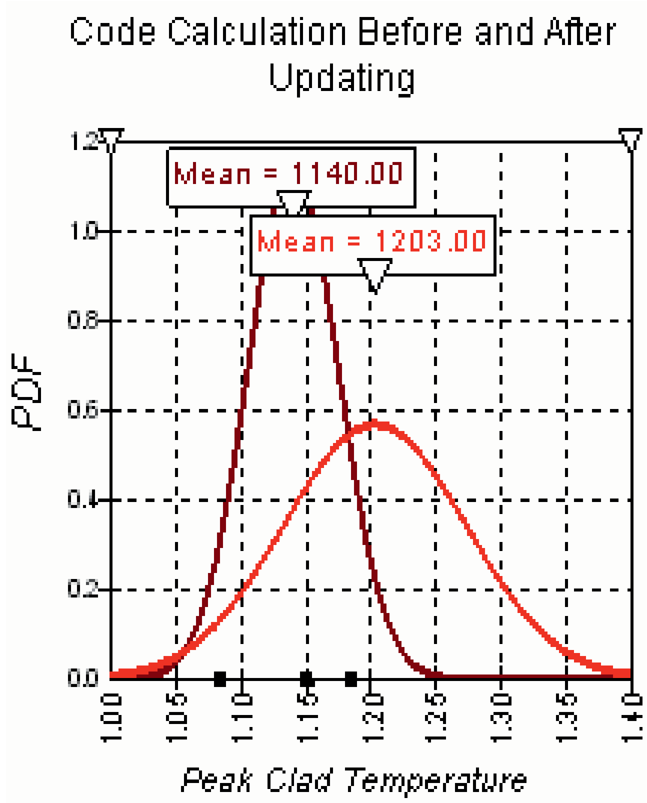
Output Updating

Code/ Test Data



Data	Mean	SD	MC Error	2.50%	Median	97.50%
Code	1140.0	35.0	0.4	1071.0	1140.0	1208.0
Experiment	1120.0	70.0	0.8	981.6	1119.0	1256.0



Concluding Remarks

- Utilization of most available data and information to include important sources of uncertainty
- Structure of models and sub-models important contributor to final result
- Depending on different conditions and availability of information and data different strategies for treating several classes of model (code structure) uncertainties proposed
- Treatment of cases involving alternative models.
- A Bayesian updating proposed for single model structure uncertainty assessment, while other techniques such as mixing, switching, maximization /minimization were proposed for alternative models.
- Output Bayesian updating proposed to account for User Errors, Numerical Approximations, Unknown and Not Considered Sources of Uncertainties (Screened input and/or Incompleteness)

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GRUPPO DI RICERCA NUCLEARE – SAN PIERO A GRADO (GRNSPG)

Gruppo Ricerca Nucleare San Piero a Grado

OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluation

Barcelona, Spain, 16-18 November 2011

Supporting Database for Uncertainty Evaluation

Title	Supporting Database for Uncertainty Evaluation	
Lecturer	F. Veronese	
Authors	A. Petruzzi, F. Fiori, A. Kovtonyuk, O. Lisovyy, F. D'Auria	
Revision/Date	1	11/01/2011

CONTENTS

INTRODUCTION

- Importance of an experimental qualified database (for assessment and uncertainty)
- Qualified experimental database is envisaged by IAEA (SRS N° 23)

↓

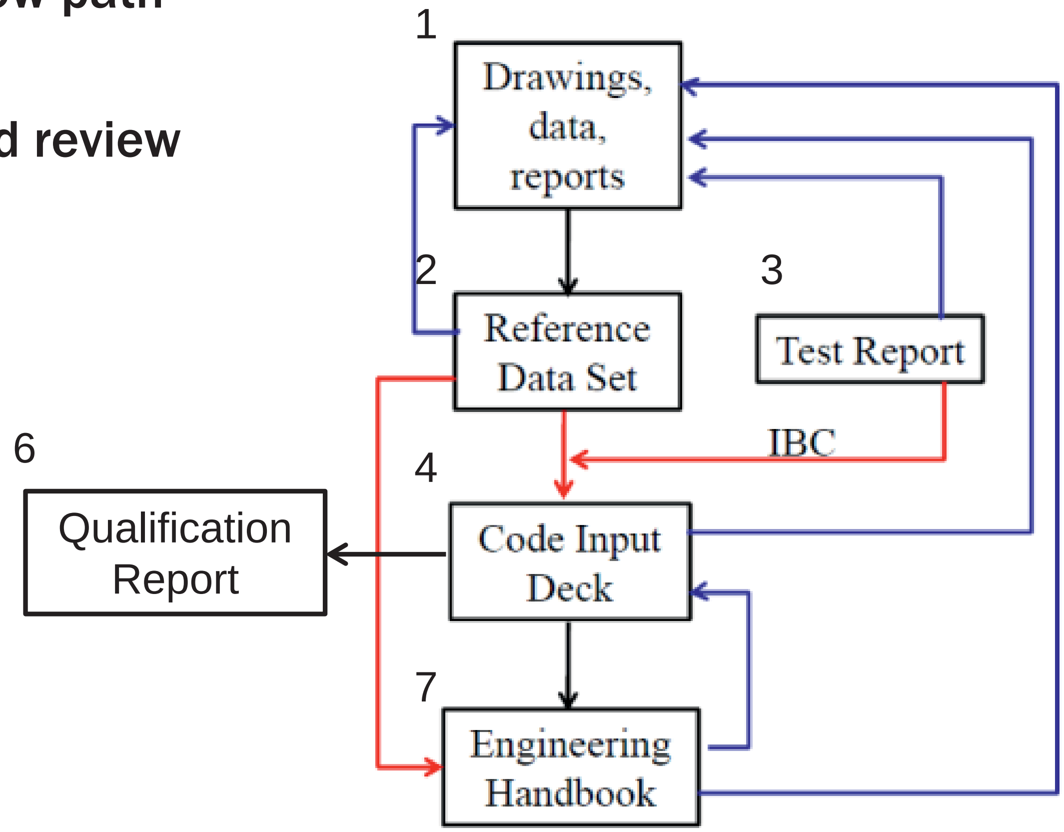
RDS, QR & EH set of document that answer the IAEA requirement

- OECD/CSNI database, ITF and STF. Widely used for V&V activities

INTRODUCTION

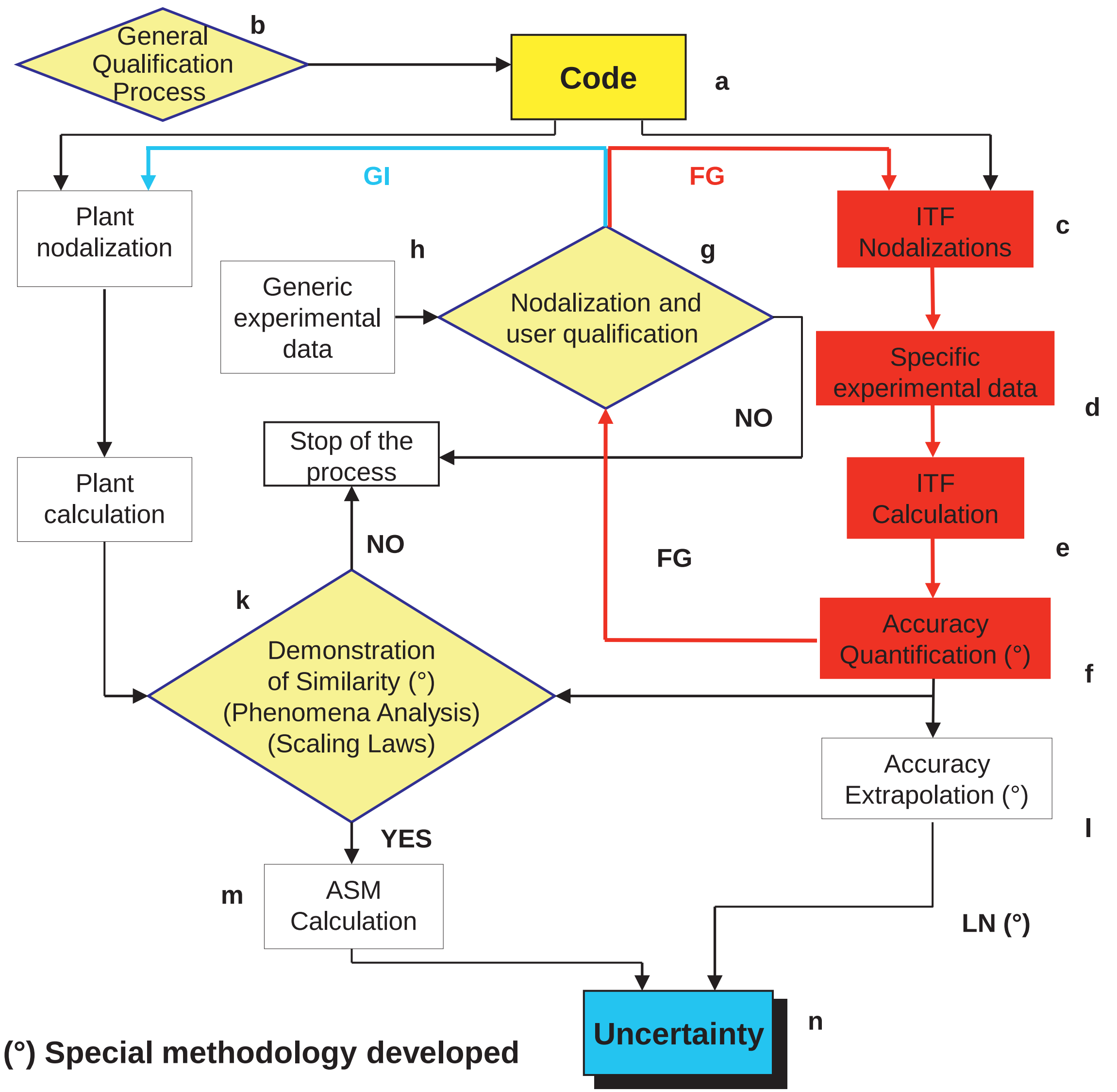
- Coherent and logic flow path
- Iterative procedure,
- Multiple feedback and review

High Quality



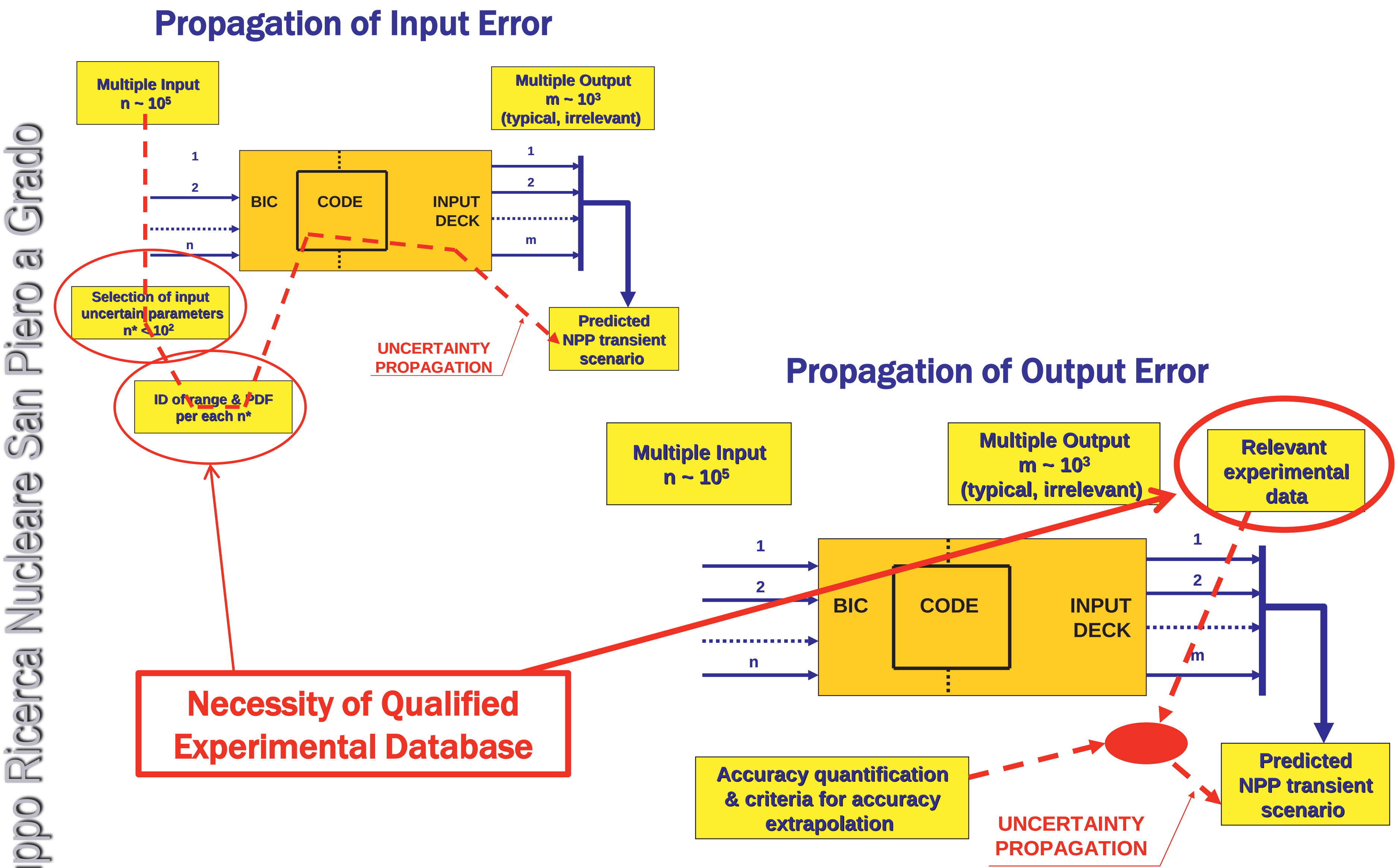
→ Feedback for review

SUPPORT TO THE UMAE



(°) Special methodology developed

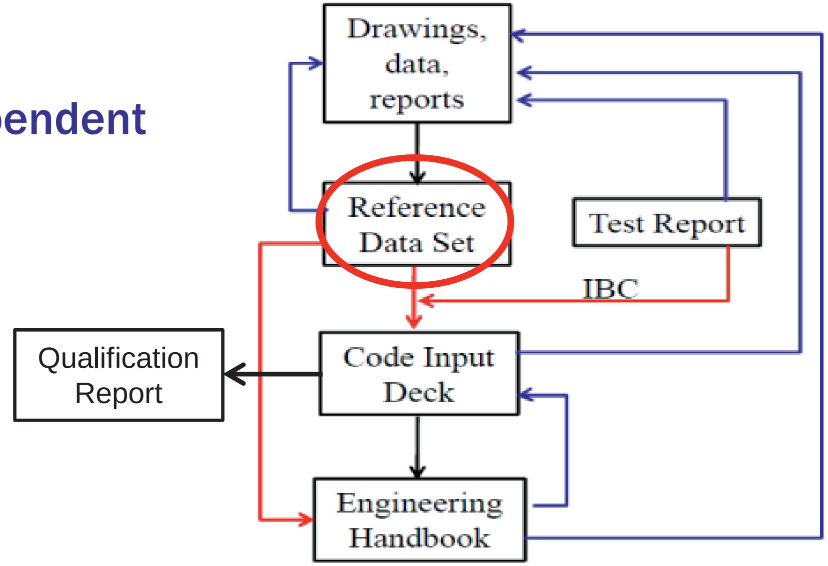
APPROACH FOR UNCERTAINTY ANALYSIS



REFERENCE DATA SET

Introduction

- IAEA guidelines (IAEA, SRS n° 23) :
  - Checking the quality of input data
  - Resolving the contradictions coming out from data
  - Explaining information on geometry, thermal and hydraulic properties
  - Performing an independent review
  - Carrying out a quality control of the database by means of relevant quality assurance procedures
  - Developing a database in a code independent form



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REFERENCE DATA SET

Purpose

- The goal of the RDS is to analyze the available documentation and to solve the possible contradictions coming out from different reports in order to produce a consistent and homogeneous set of data of the facility
  - Different facility modifications may have occured during the entire duration of the experimental campaign
- The RDS data are available for input qualification and input development

REFERENCE DATA SET

Structure and Sample, RDS facility

- The RDS realated with the design of a facility may consist of the following sections:
  - Layout of the facility
  - Collection of geometrical data (length, volumes, areas, elevations) for each subsystem and component of the facility
  - Collection of specific data for complex component (pumps, valves, heaters, etc...)
  - Identification of geometrical discontinuities and evaluation of pressure loss coefficients (normal operation)
  - Material properties
  - Measurement system
  - Nominal heat losses
  - Nuclear data

REFERENCE DATA SET

Structure and Sample, RDS facility

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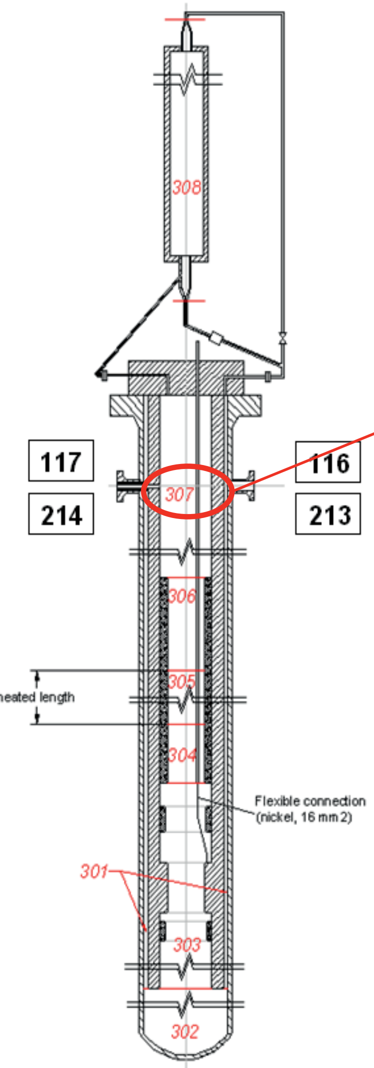


Module description

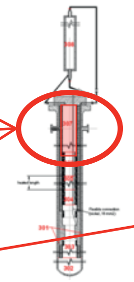
Module position

Geometrical description

Lengths

Module number

Areas

Pressure vessel				
Module number	307			
Description	Upper plenum, part 2			Vertical cylinder
Parameters	**Evaluation**	**Value**		**Remarks**
Outside diameter	D_{o1} = 10.75 mm (rod bundle) D_{o2} = 212 mm (inside wall of barrel honeycomb) Do3 = 420 mm (upper plate)	$1.075 \cdot 10^{-2}$ $2.12 \cdot 10^{-1}$ $4.2 \cdot 10^{-1}$	m	Draw. 13
Inside diameter	D_i = 198 mm	$1.98 \cdot 10^{-1}$	m	
Number of rod bundle	64			Draw. 13
Length	L_1 = 2015 + 315 - 328 = 2002 mm L_2 = 130 mm (upper plate)	2.002 $1.3 \cdot 10^{-1}$	m	Draw. 13
Outside radius	R_{o1} = 10.75/2 = 5.37 mm R_{o2} = 212/2 = 106 mm R_{o3} = 420/2 = 210 mm	$5.37 \cdot 10^{-3}$ $1.06 \cdot 10^{-1}$ $2.1 \cdot 10^{-1}$	m	
Inside radius	R_i = 198/2 = 99 mm	$9.9 \cdot 10^{-2}$	m	
Elevation change	ΔH = 2002 mm	2.002	m	
Flow area	$A_{f1} = \pi \cdot R_i^2 = \pi \cdot 99^2 = 30791\ mm^2$ $A_{f2} = \pi \cdot R_{o1}^2 = \pi \cdot 5.37^2 = 90.6$ $A_f = A_{f1} - 64 \cdot A_{f2} = 30791 - 64 \cdot 90.6$ $= 24993\ mm^2$	$2.4993 \cdot 10^{-2}$	m^2	Draw. 13
Inside surface area	**Heat exchange with inside wall of barrel honeycomb** $S_i = 2\pi \cdot R_i \cdot L_1 = 2\pi \cdot 99 \cdot 2002$ $= 1245315\ mm^2$	1.245315	m^2	Draw. 13
Outside surface area	**Outer surface of inside wall of barrel honeycomb barrel** $S_o = 2\pi \cdot R_{o2} \cdot L_1$ $= 2\pi \cdot 106 \cdot 2002 = 1346688\ mm^2$	1.346688	m^2	

(to be continued)

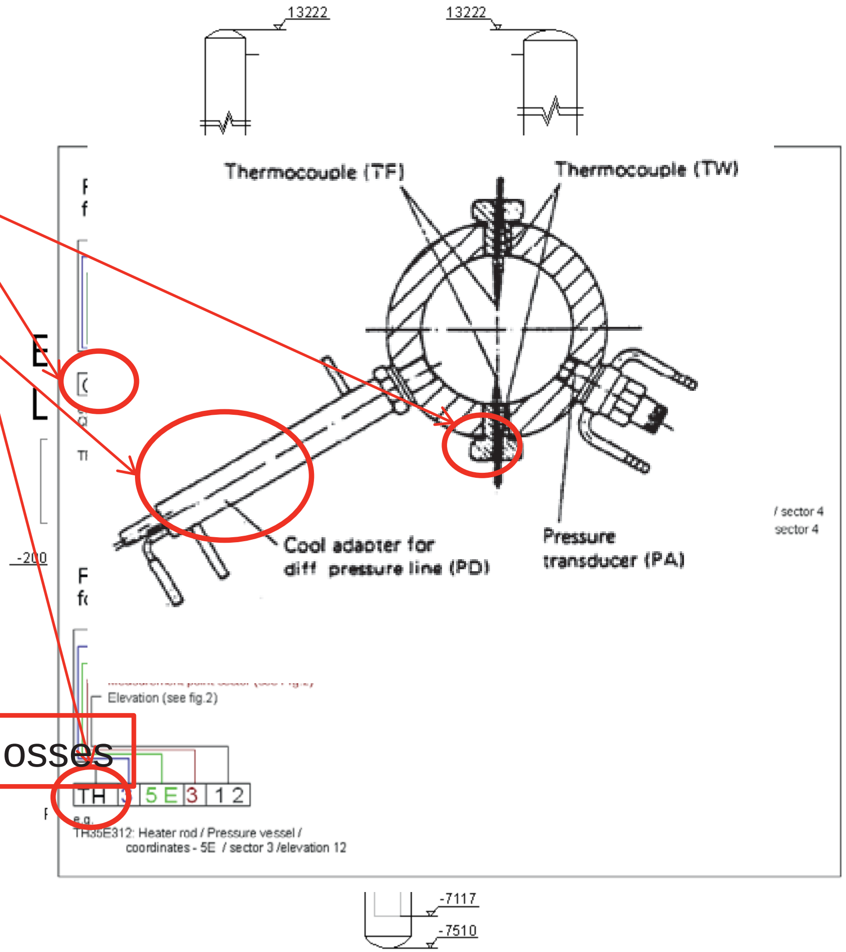
REFERENCE DATA SET

Structure and Sample, RDS facility

Geometry variation and measurement inserts introduce pressure losses in the system

- Identify measured parameters
- Identify measurement locations
- Classify measurement insert types

Evaluation of pressure losses



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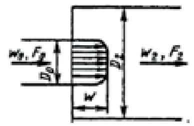
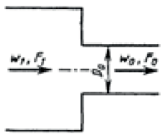
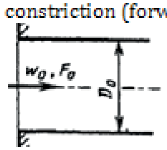
REFERENCE DATA SET

Structure and Sample, RDS facility

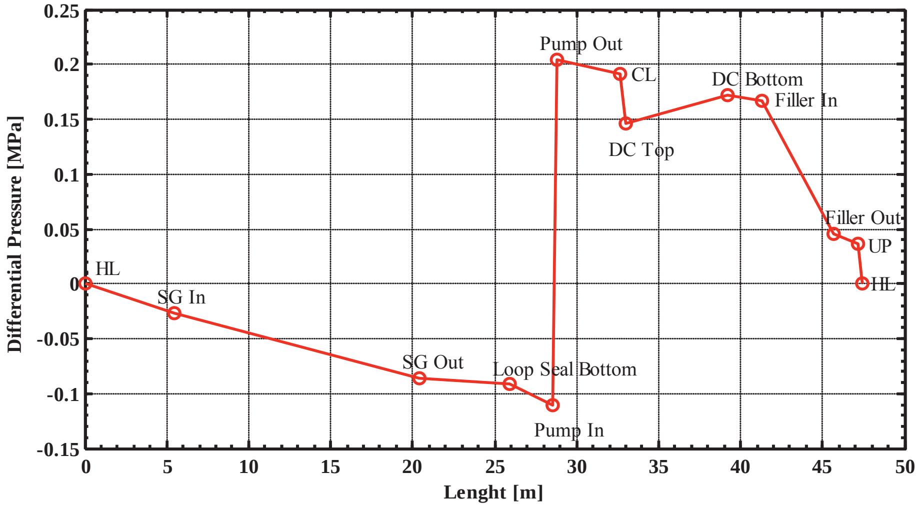
- Modules number
- Geometrical configurations
- Parameters values and adopted formulas
- K-loss coefficients
- References

Nº	Element of system	Parameters	G_i kg/s	t °C	$\nu_i \cdot 10^{-7}$ m²/s	ρ kg/m³	w_i m/s	Re·10⁶	Evaluation	k_{loc}	Remarks
306-307	Expansion (forward) 	$\frac{F_0}{F_2} = \frac{8115}{24982} = 0.325$	28	326	1.26	654	5.29	4.26	$k_{loc} = \left(1 - \frac{F_0}{F_2}\right)^2$	0.452	Ref. [2] Sec. 4-1 (p. 146, 158)
	Expansion (reverse) 	$\frac{F_0}{F_1} = \frac{8115}{24982} = 0.325$	28	326	1.26	654	5.29	4.26	$k_{loc} = 0.5\left(1 - \frac{F_0}{F_1}\right)^{\frac{3}{4}}$	0.371	Ref. [2] Sec. 4-9 (p. 151, 165)
307-308	Outlet from upper plenum: constriction (forward) 	$\frac{B}{D_r} = 0$ $\frac{\delta_1}{D_r} = 0$	0.4	294	1.26	740	0.6	0.16		0.5	Ref. [2] Sec. 3-1 (p. 122)
	Outlet from upper plenum: constriction (reverse)		0.4	294	1.26	740	0.6	0.16		1	Ref. [2] Sec. 11-1 p. 510

REFERENCE DATA SET

Structure and Sample, RDS test

PD Vs length



Sequence of Events

Events	Time (s)
Break valve starts to open, blowdown initiated	0
Primary system pressure equal 132 bar (core heating power and secondary system isolation and cooldown trip signals enabled)	1.8
Feedwater valves and steam valve at condenser inlet start to close Cooldown of secondary system initiated	2.0
Saturation in hot legs	2.5
Break valve fully open Core heating power decay starts	3.0
Feedwater valves and steam valve condenser inlet fully closed	3.5
Primary system pressure equal 117 bar (HPIS trip signal enabled)	5.4
Primary system pressure equal 110 bar (MCPs trip signal enabled)	6.7
Main coolant pumps coastdown initiated	8.0
Saturation in cold legs	15.8
PRZ surge line uncovers	21.0
Saturation in lower plenum	31.0
HPIS water injection initiated	41.0

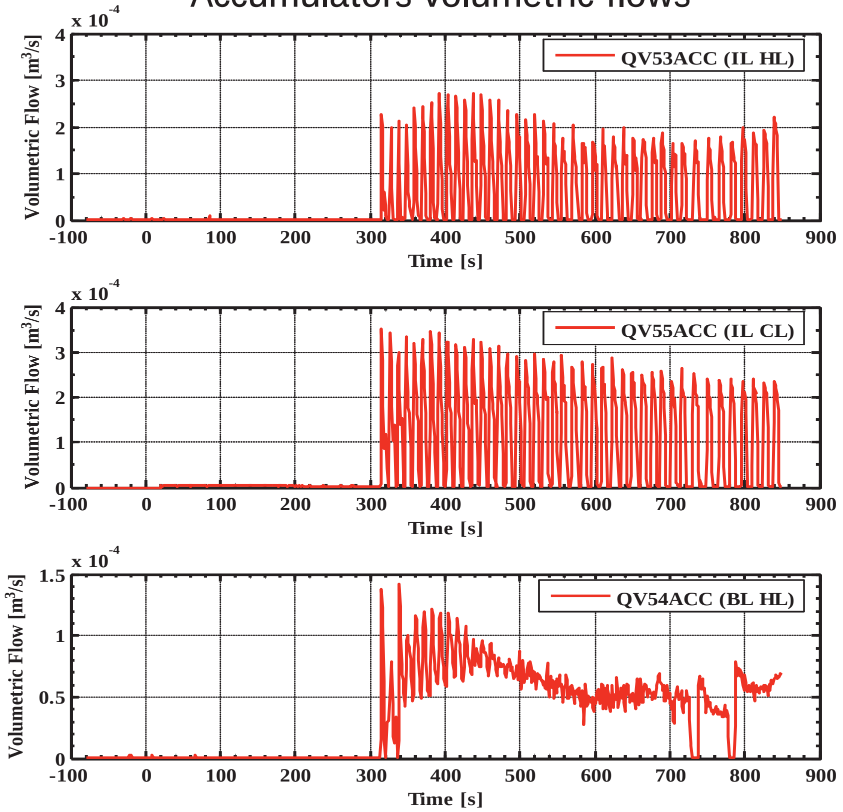
Relevant ICs

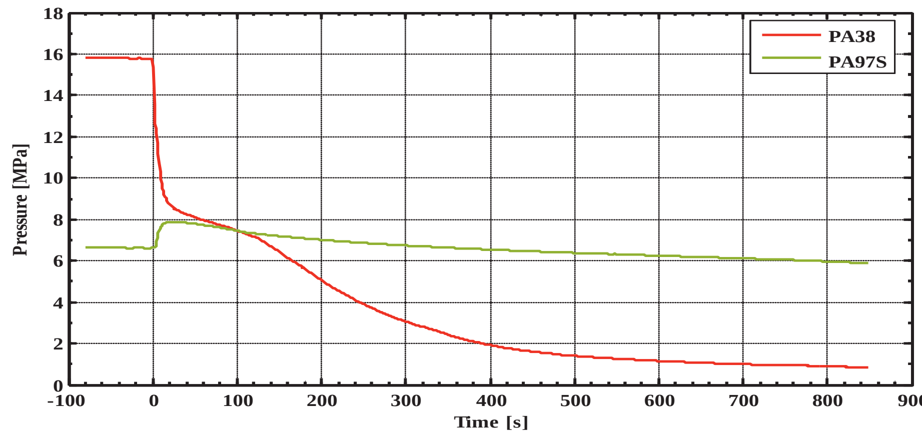
Parameters	Location	Value
Primary System:		
Mass Flow	Intact loop Broken loop	20.8 kg/s 6.7 kg/s
Pressure	Upper plenum	15.8 MPa
Fluid Temperature	Vessel outlet • Intact loop • Broken loop Vessel inlet • Intact loop • Broken loop	 327.9 °C 327.8 °C 296.2 °C 295.4 °C
Water Level	Pressurizer	c. 5.2 m
Temperature	Pressurizer	346 °C
Power	Core	5.20 MW
Water Volume	Accumulator • Intact loop • Broken loop	 246 l 76 l
Gas Volume	Accumulator • Intact loop • Broken loop	 34 l 18 l
Temperature	Accumulator • Intact loop • Broken loop	 c. 30 °C c. 30 °C
Mass Flow	MCP seal water injection Intact loop Broken loop	 0.01 kg/s 0.0087 kg/s
Temperature	MCP seal water injection	c. 30 °C
Water Temperature	HPIS	28 °C
Secondary System:		
Mass Flow	Steam generator Intact loop Broken loop	 2.0 kg/s 0.66 kg/s
Pressure	Steam dome Intact Loop Broken Loop	 6.62 MPa 6.62 MPa

REFERENCE DATA SET

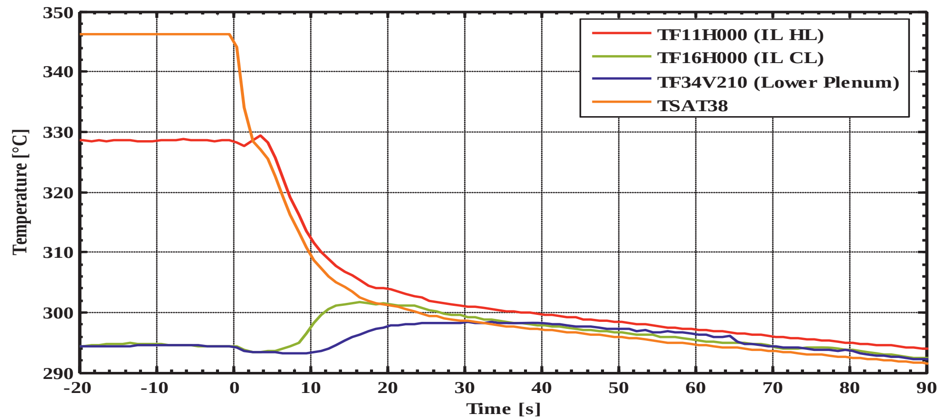
Structure and Sample, RDS test

Accumulators volumetric flows





Feedwater volumetric flow (short time)

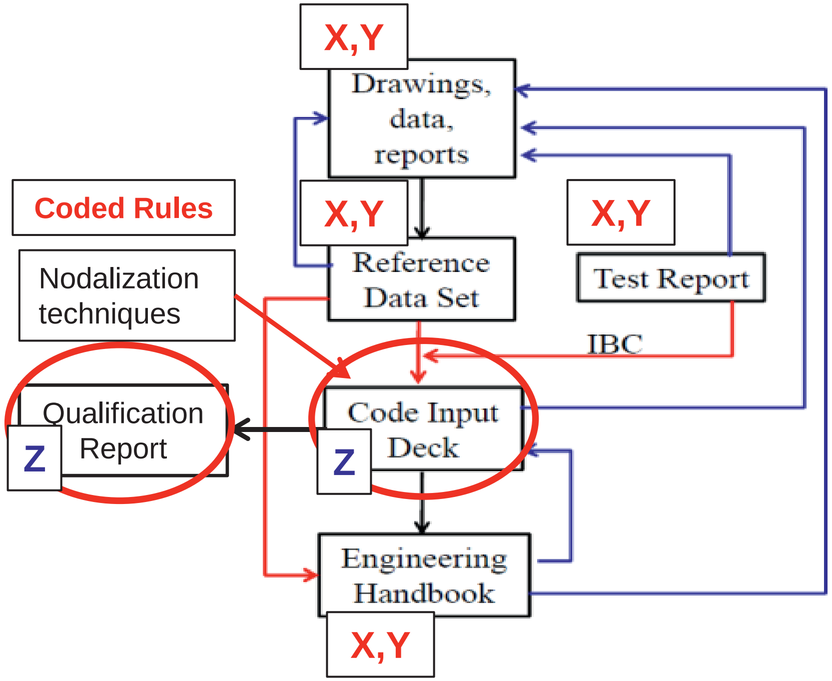


Core power (short time)

INPUT DECK DEVELOPMENT&QUALIFICATION

Introduction

- Nodalization preparation: main choices of the model characteristics and preliminary code resources distribution (data from RDS)
- Nodalization schematization according to the pre-set nodalization strategies
- Input writing following a pre-set structure



X,Y, Z: indicate three different analysts

- The Qualification Report (QR) collects the results of the qualification procedures of the code input

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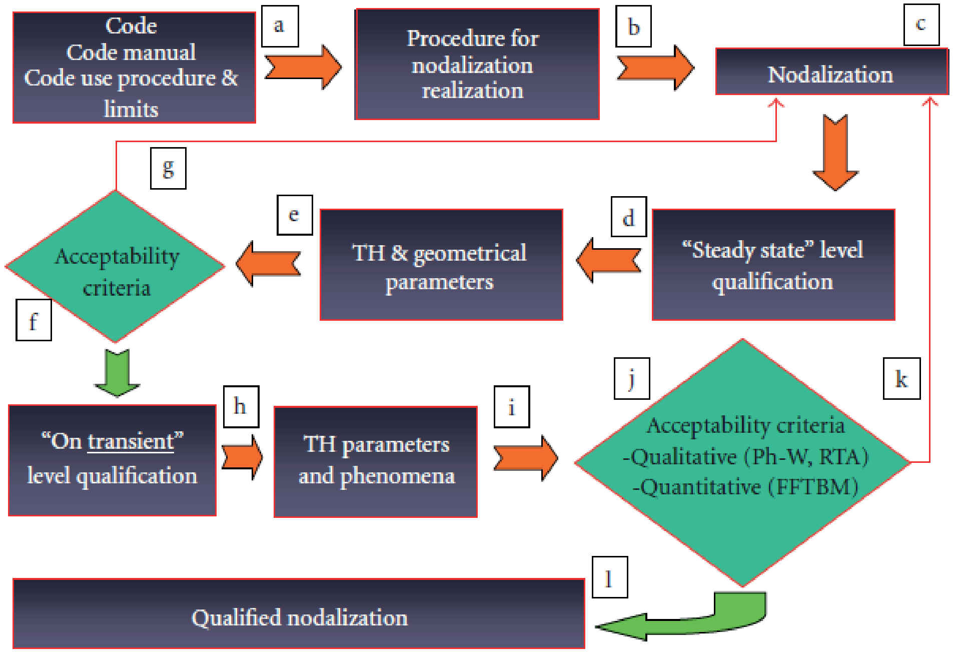
INPUT DECK DEVELOPMENT&QUALIFICATION

Structure and Sample

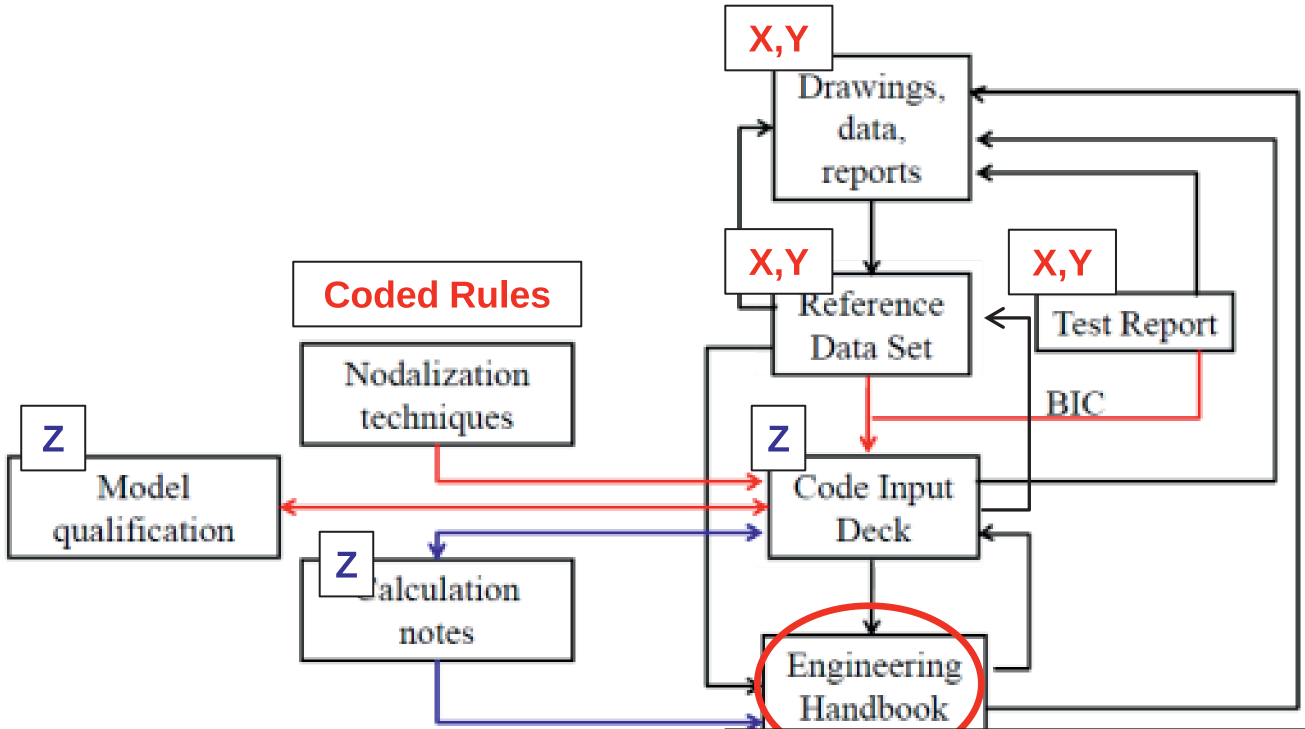
- QR to demonstrate that code results are qualitative and quantitative acceptable with respect to fixed acceptance criteria. QR should contain:
  - Demonstration of geometrical fidelity
  - Qualification at steady-state level
  - Qualification at transient level (both qualitative and quantitative)



ENGINEERING HANDBOOK

Introduction

- Final step of the process to set up a qualified database, IAEA states that a: *“documents contains a full description of how the database has been converted into an input data deck for a specific computer code”*, (IAEA, SRS n°23) should be available



X,Y, Z: indicate three different analysts

1° X,Y, input entries documentation
2° Z, rationale & user choices

ENGINEERING HANDBOOK

Introduction

- EH constitute the technical rational for the input, providing engineering justification of the adopted assumption and summarize the model's input file
- Make the use of the input by a third user easier, preventing errors and misunderstanding
- It is set up on only after the nodalization is qualified and frozen
- EH shall contains:
  - Methods and assumptions used to convert the RDS information into the code input data
  - All the transliteration of the calculation notes (traceability of the information)
  - Nodalization schemes of the components
  - Adequate description and explanation of all adopted modeling assumptions

FINAL STEP TO SET UP A QUALIFIED EXP DATABASE

(review of the input deck and of the RDS)

ENGINEERING HANDBOOK

Structure & Samples

❑ R5-3D© nodalization description

General Zone	Zone	Name	Number	Type	Document Section
Primary Side					
Intact Loop	IL HL	ILHL-1	100	BRANCH	2.2.3.1
		ILHL-2	105	BRANCH	
		ILHL-3	110	PIPE	
	IL SG	ILSG-IN	115	BRANCH	2.2.3.2
		IL-UT	120	PIPE	
		ILSG-OUT	125	BRANCH	
	IL LOOP SEAL	ILLS-1	130	PIPE	2.2.3.3
	IL PUMP	IL-PUMP	140	PUMP	2.2.3.4
	IL CL	ILCL-1	150	PIPE	2.2.3.5
		ILCL-2	160	BRANCH	
		ILCL-3	170	BRANCH	
Broken Loop	BL HL	BLHL-1	200	BRANCH	2.2.5.1
		BLHL-2	205	BRANCH	
		BLHL-3	210	PIPE	
	BL SG	BLSG-IN	215	BRANCH	2.2.5.2
		BL-UT	220	PIPE	
		BLSG-OUT	225	BRANCH	
	BL LOOP SEAL	BLLS-1	230	PIPE	2.2.5.3
	BL PUMP	BL-PUMP	240	PUMP	2.2.5.4
	BL CL	BLCL-1	250	SNGLVOL	
		BL-ROTOR	251	VALVE	2.2.5.5
		BLCL-2	255	PIPE	2.2.5.6
		BLCL2	256	SNGLJUN	
		BLCL-3	260	PIPE	
		BLCL-4	265	BRANCH	
		BLCL-5	270	BRANCH	

Link to the document section (component by component)

User friendly

Structure & Samples

Vessel	DOWNCOMER	DC-1	300	PIPE	2.2.1.1
		DC-2	305	BRANCH	
	LOWER PLENUM	LP-1	310	BRANCH	2.2.1.2
		LP-2	315	PIPE	
	CORE	CORE-B	325	BRANCH	2.2.1.3
		CORE-A	330	PIPE	
		CORE-T	335	BRANCH	
	UP	UP-1	340	PIPE	2.2.1.4
		UP-2	345	BRANCH	
		UP-3	350	BRANCH	
Upper Head	UH	UH-1	370	PIPE	2.2.2.5
		UH-P1	372	BRANCH	
		UH-UP	375	PIPE	
		UH-DC1	381	PIPE	
		UH-DC1	383	SNGLJUN	
		UH-DC2	384	SNGLJUN	
		UH-DC2	386	PIPE	
		UH-DC2	388	VALVE	
		UH-DC3	389	PIPE	
PRZ	IL SURGE LINE	PRZ-SL1	400	PIPE	2.2.6
	BL SURGE LINE	PRZ-SL2	405	PIPE	
	SURGE LINE	PRZ-SL3	410	BRANCH	
		PRZ-SL4	415	PIPE	
	PRZ VESSEL	PRZ-BOT1	420	BRANCH	
		PRZ-BOT2	425	BRANCH	
		PRZ-CYL	430	PIPE	
		PRZ-TOP	440	BRANCH	

(to be continued)

ENGINEERING HANDBOOK

Structure & Samples

General Zone	Zone	Name	Number	Type	Document Section
Secondary Side					
IL SG	DOWNCOMER	ILSG-DC1	600	PIPE	2.3.1.1
		ILSG-DC2	601	BRANCH	
	RISER	ILSG-RSR	605	PIPE	2.3.1.2
	SEPARATOR and STEAM DOME	ILSG-SEP	610	SEPARATR	2.3.1.3
		ILSG-DOM	620	BRANCH	
	IL SG ANNULUS	ILSG-AN1	630	BRANCH	
		ILSG-AN2	635	BRANCH	
BL SG	DOWNCOMER	BLSG-SC1	700	PIPE	2.3.3.1
		BLSG-DC2	701	BRANCH	
	RISER	BLSG-RSR	705	PIPE	2.3.3.2
	SEPARATOR and STEAM DOME	BLSG-SEP	710	SEPARATR	2.3.3.3
		BLSG-DOM	720	BRANCH	
	BL SG ANNULUS	BL SG-AN1	730	BRANCH	
		BL SG-AN2	735	BRANCH	
Primary Side Boundary Conditions					
IL Pump	Seal Water	il.pu.st	180	TMDPVOL	2.2.10.2
		il.pusx	181	TMDPJUN	
BL Pump	Seal Water	bl.pu.st	280	TMDPVOL	2.2.10.3
		bl.pusx	281	TMDPJUN	
Pump	Seal Water drainage	pe.s.exj	398	TMDPJUN	2.2.10.4
		pu.s.exv	399	TMDPVOL	

ENGINEERING HANDBOOK

Structure & Samples

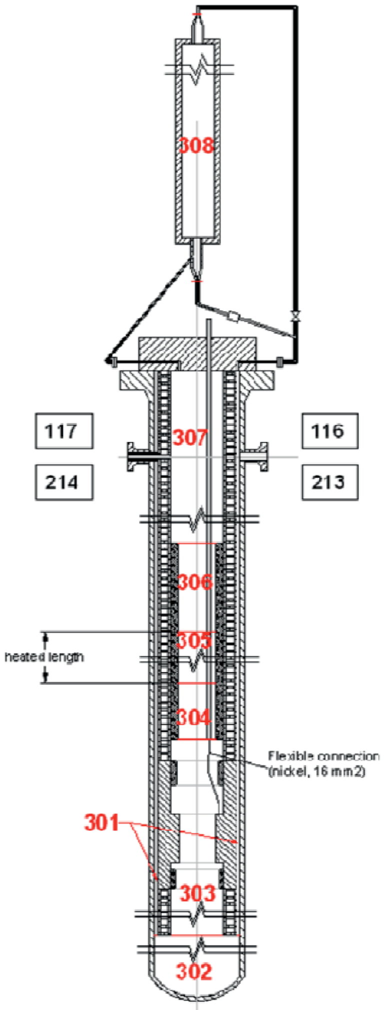
Secondary Side Boundary Conditions					
IL SG P Control	-	il.sgcx	660	VALVE	2.4.1
	-	Il.sg.v	661	TMDPVOL	
BL SG P Control	-	bl.sgcx	760	VALVE	2.4.2
	-	bl.sg.v	761	TMDPVOL	
IL SG Cooldown	-	isg-cool	680	VALVE	2.4.3
			681	TMDPVOL	
BL SG Cooldown	-	bsg-cool	780	VALVE	2.4.4
	-		781	TMDPVOL	
IL SG Main Feedwater	Feedwater tank	sg.fw.ta	685	TMDPVOL	2.4.5
	Feedwater Main	IL-MFW	686	TMDPJUN	
BL SG Main Feedwater	Feedwater tank	sg.fw.ta	785	TMDPVOL	2.4.6
	Feedwater Main	BL-MFW	786	TMDPJUN	
ECCS					
IL ACCUM	IL ACC TANK	IL-ACC	800	ACCUM	2.2.9.2
	ACC LINE	ILACC-L1	805	PIPE	
		ILCL-ACC	810	VALVE	
		ILACC-CL	812	PIPE	
		ILACC-CL	814	BRANCH	
		ILACC-CL	815	PIPE	
		ILHL-ACC	820	VALVE	
		ILACC-HL	822	PIPE	
		ILACC-HL	824	BRANCH	
		ILACC-HL	825	PIPE	

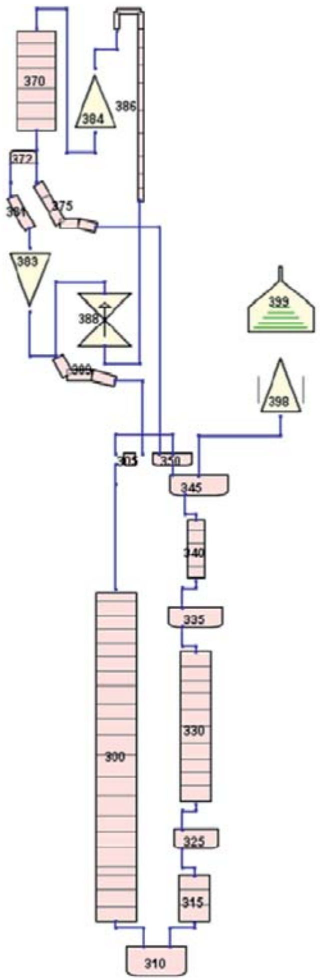
(to be continued)

ENGINEERING HANDBOOK

Structure & Samples

- RPV (from RDS) reference for the data used for the input

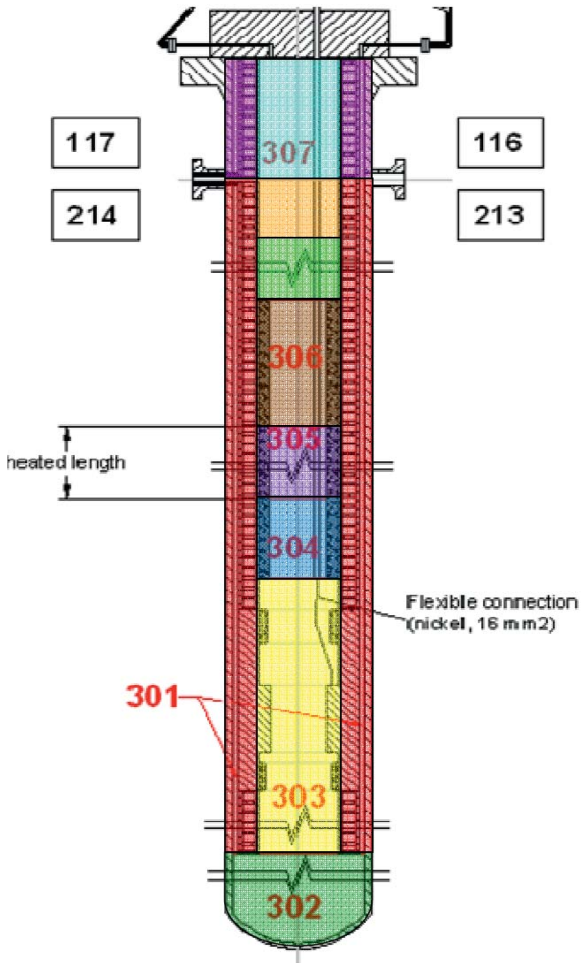


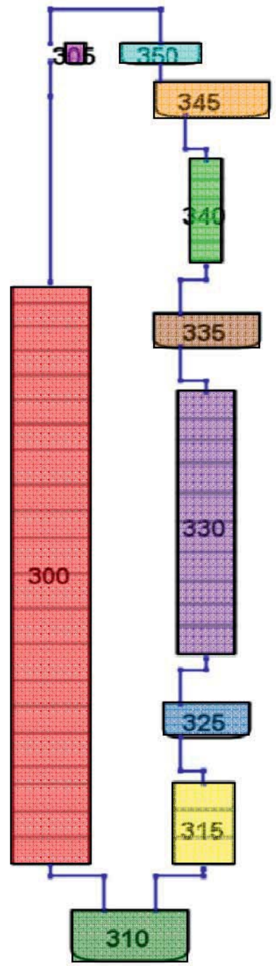


ENGINEERING HANDBOOK

Structure & Samples

- RPV (from RDS) reference for the data used for the input





2.2.1.2 Lower Plenum

HYDRO COMPONENTS

- **Rationale**
- **User choices**
- **Models (flag)**
  - BRANCH 310 (HEMISPHERICAL HEAD)

  Default

  - PIPE 315 (CORE INLET)

  Default
- **Geometry Data**
  - BRANCH 310 (HEMISPHERICAL HEAD)

Component 310 models the bottom of the reactor pressure vessel, it consists of the cylindrical part and the hemispherical bottom. The internal diameter of the core vessel is 0.312 m, the internal radius of the hemispherical bottom is 0.373 m and the average flow area of the component is $7.0954 \cdot 10^{-2}$ m^2 (*Equation 2-7*). See *Table 2-2* for detailed geometry summary.

$$A_{310} = V_{M302} / L_{M302} = 0.026465 \ / \ 0.373 = 7.0954 \cdot 10^{-2} \ m^2 \qquad \text{Equation 2-4}$$

- PIPE 315 (CORE INLET)

Component 315 models the entrance region to the core barrel. It is subdivided in 3 cells, the total length is 1.006 m and the average flow area is $2.5651 \cdot 10^{-2}$ m^2 (*Equation 2-5*). The component PIPE 315 corresponds to module 303. See *Table 2-2* for detailed geometry summary.

$$A_{315} = V_{M303} / L_{M303} = 0.025805 \ / \ 1.006 = 2.5651 \cdot 10^{-2} \ m^2 \qquad \text{Equation 2-5}$$

- **Junction Data**
  - <u>BRANCH 310 (HEMISPHERICAL HEAD)</u>

Component 310 (bottom of the RPV) has 2 junctions. The first one connects, "outlet face", of the component 310 to the last cell (cell 21) outlet face of component 300 (downcomer). The second junction connects the "outlet face" of component 310 to the first cell, "outlet face", of component 315 (lower plenum).

  - <u>PIPE 315 (CORE INLET)</u>

No special model is used for the internal junctions. The junction flow area is not specified. Forward K-loss coefficient of 2.202 and a reverse K-loss coefficient of 2.221 are applied to junction number 2 connecting cells 31502 and 31503. See *Table* 2-3 for detailed junction summary.

Structure & Samples

Comp. Name	Comp. N°	Volume N°	Comp. Type	Length	Area	Volume	D_{hd}	Elevation change	Angle	Outlet Elevation	Wall Roughness	Volume Control Flag tlpvbfe	Comment
DC-1	300	1	PIPE	0.200	$1.131\cdot10^{-2}$	-	0.024	-0.200	-90°		$4.5\cdot10^{-5}$	0000000	
		2		0.287				-0.287					
		3		0.300				-0.300					
		4		0.300				-0.300					
		5		0.300				-0.300					
		6		0.300				-0.300					
		7		0.328				-0.328					
		8		0.412				-0.412					
		9		0.331				-0.331					
		10		0.332				-0.332					
		11		0.437				-0.437					
		12		0.438				-0.438					
		13		0.438				-0.438					
		14		0.437				-0.437					
		15		0.332				-0.332					
		16		0.331				-0.331					
		17		0.412				-0.412					
		18		0.200				-0.200					
		19		0.335				-0.335					
		20		0.335				-0.335					
		21		0.336				-0.336					
DC-2	305	1	BRANCH	0.315	$1.131\cdot10^{-2}$	-	0.024	0.315	90°		$4.0\cdot10^{-5}$	0000000	
LP-1	310	1	BRANCH	0.373	-	$2.647\cdot10^{-2}$	0.024	0.315	90°		$4.0\cdot10^{-5}$	0000000	
LP-2	315	1	PIPE	0.336	$2.565\cdot10^{-2}$	-		0.336	90°		$4.0\cdot10^{-5}$	0000000	
		2		0.335				0.335					
		3		0.335				0.335					
CORE-B	325	1	BRANCH	0.200	$8.126\cdot10^{-3}$			0.200	90°		$4.0\cdot10^{-5}$	0000000	
CORE-A	330	1	PIPE	0.412	$8.1152\cdot10^{-3}$	-	-	0.412	90°		$4.0\cdot10^{-5}$	0000100	Squared cross-section
		2		0.331				0.331					
		3		0.332				0.332					
		4		0.437				0.437					
		5		0.438				0.438					
		6		0.438				0.438					
		7		0.437				0.437					
		8		0.332				0.332					
		9		0.331				0.331					
		10		0.412				0.412					

(to be continued)

Structure & Samples

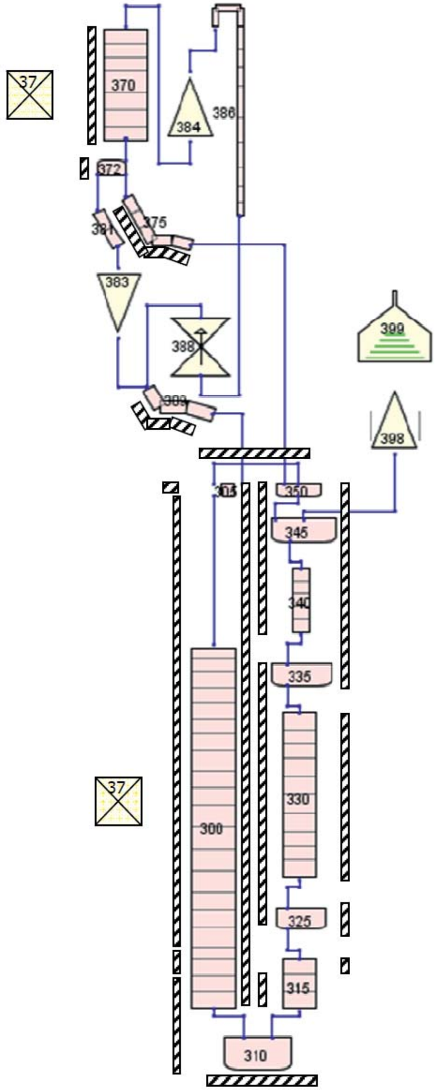
Component Name	Comp. N°	Component Type	Junction Number	From Component	To Component	Junction Area	Junction Flag jefvcahs	Loss Coefficient K_f	Loss Coefficient K_r	Description
DC-1	300	PIPE	1	30001	30002	-	00000000	0	0	
			2	30002	30003					
			3	30003	30004					
			4	30004	30005					
			5	30005	30006					
			6	30006	30007					
			7	30007	30008					
			8	30008	30009					
			9	30009	30010					
			10	30010	30011					
			11	30011	30012					
			12	30012	30013					
			13	30013	30014					
			14	30014	30015					
			15	30015	30016					
			16	30016	30017					
			17	30017	30018					
			18	30018	30019					
			19	30019	30020					
			20	30020	30021					
DC-2	305	BRANCH	1	300010001	305010001	-	00000000	0.000	0.000	
			2	305010002	350010002	$3.927 \cdot 10^{-5}$		4.500	4.500	
			3	305010002	389030002	$3.142 \cdot 10^{-4}$		1.669	1.669	
LP-1	310	BRANCH	1	300210002	310010002	-	00000000	0.723	0.443	
			2	310010002	315010001			0.341	0.360	
LP-2	315	PIPE	1	31501	31502	-	00000000	0.000	0.000	
			2	31502	31503			2.200	2.200	
CORE-B	325	BRANCH	1	315030002	325010001	-	00000000	0.393	0.526	
			2	325010002	330010001	-		0.220	0.220	

(to be continued)

ENGINEERING HANDBOOK

Structure & Samples

❑ Heat Structure



2.2.2.2 *Lower Plenum Heat Structure*

The lower plenum structures are modelled by heat structures 3101, 3151 and part of 3001 (see description in 2.2.2.1).

- **Rationale**
- **User choices**
- **Models (flag)**
  - Pressure Vessel, LP Hemispherical Head (3101)
  - Lower Plenum Barrel (3151)
- **Calculation notes: Geometry Data**
  - Pressure Vessel, LP Hemispherical Head (3101)

The heat structure 3101 models the hemispherical bottom of the pressure vessel, it is of rectangular geometry type. The left side of the heat structure is connected with hydrodynamic component 310, the heat transfer option used is 101. The right surface is connected to the hydrodynamic component 30 which represents the environment. The geometry of the component is taken from RDS module 301: the wall thickness of the HS is $t = 0.0158\ \mathrm{m}$. The HS heat exchange area is 0.0862 m^2. For more detailed HS information see *Table 2-5, Table 2-6* and *Table 2-7.*

- Lower Plenum Barrel (3151)

The heat structure 3151 represents the lower plenum of the barrel. The HS has cylindrical geometry and it comprise three axial heat structure of the same geometry. The left side of the structure is connected to the hydrodynamic component 315, cells one and two. The inner and outer diameters of the HS are taken from RDS module 303: $\mathrm{D_{in}} = 0.0909\ \mathrm{m}$ and $\mathrm{D_{out}} = 0.212\ \mathrm{m}$ (*Equation 2-33*). HS has a total length of 0.671 m ($L_{3151} = L_{310(1\div2)}$). For more detailed HS information see *Table 2-5, Table 2-6* and *Table 2-7.*

$$V_1 = \frac{\pi}{4} \cdot (0.212^2 - 0.198^2) \cdot 0.43 \qquad \text{Equation 2-24}$$

$$V_2 = \frac{\pi}{4} \cdot (0.212^2 - 0.15^2) \cdot 0.065 \qquad \text{Equation 2-25}$$

$$V_3 = \frac{\pi}{4} \cdot (0.212^2 - 0.198^2) \cdot 0.025 \qquad \text{Equation 2-26}$$

$$V_4 = \frac{\pi}{4} \cdot (0.212^2 - 0.198^2) \cdot 0.16 \qquad \text{Equation 2-27}$$

$$V_5 = \frac{\pi}{4} \cdot (0.212^2 - 0.198^2) \cdot 0.136 \qquad \text{Equation 2-28}$$

$$V_6 = \frac{\pi}{4} * (0.212^2 - 0.15^2) \cdot 0.09 \qquad \text{Equation 2-29}$$

$$V_7 = \frac{\pi}{4} \cdot (0.212^2 - 0.198^2) \cdot 0.1 \qquad \text{Equation 2-30}$$

$$V_{tot} = \sum_{i=1}^{7} V_i = 0.009372\ m^3 \qquad \text{Equation 2-31}$$

$$A_{eq} = \frac{V_{tot}}{L_{M303}} = 0.09316\ m^2 \quad \text{and} \quad A_{eq} = \frac{\pi}{4} \cdot \left(D_{out}^2 - \left(D_{in}^{ef}\right)^2\right) \qquad \text{Equation 2-32}$$

these two equations are used to calculate the average diameter of the HS:

$$D_{in}^{ef} = \sqrt{D_{out}^2 - 4 \cdot \frac{A_{eq}}{\pi}} = 0.0909\ \mathrm{m} \qquad \text{Equation 2-33}$$

2.2.2.3 *Core Heat Structure*

The core is modelled by heat structures 3150, 3250, from 3300 to 3310 and 3350. The component numbered from 3300 to 3310 model the heated length of the core, as it is discussed in the RDS the 14% of the total power is deposited outside this part of the bundle, the use of the adjective "unheated" for components 3150, 3250 and 3350 should not be misleading, as also this components are active structure, for all the HS described in the following the power is imposed in table 900 (see *Table 3-1*), different source multiplier are used to set the cosine shape power curve as specified in each experiment.

- **Rationale**
- **User choices**
- **Models (flag)**
  - Lower Unheated Region 1 (3150)
  - Lower Unheated Region 2 (3250)
  - Heater Rod Bundle Part 1 (3300)
  - Heater Rod Bundle 2 (3301)
  - Heater Rod Bundle 3 (3302)
  - Heater Rod Bundle 4 (3303)
  - Heater Rod Bundle 5 (3304)
  - Heater Rod Bundle 6 (3305)
  - Heater Rod Bundle 7 (3306)
  - Heater Rod Bundle 8 (3307)
  - Heater Rod Bundle 9 (3308)
  - Heater Rod Bundle 10 (3309)
  - Core Filler (3310)
  - Upper Unheated Part (3350)
- **Calculation notes: Geometry Data**
  - Lower Unheated Region 1 (3150)

The Heat Structure 3150 represents the lower unheated part of the heated roads (the nickel flexible connection).The geometry used for this HS is cylindrical, and it is composed of only one axial HS. A symmetric boundary condition is applied on the left side, that for this structure correspond to the axis. Its right side is connected with hydrodynamic component 315, the heat transfer model applied is 110. The outer diameter of the HS is taken from RDS module 303: $\mathrm{D_{out}} = 0.0045\ \mathrm{m}$

3 GENERAL TABLE

3.1 Core Power table

The steady state power for each test is imposed on card 20290000 (see *Table 3-1*). General table 900 is used to imposed the decay power as specified for each experiment. The table is activated with the trip 1900 (see section 5.3.1.6). The entries of *Table 3-1* may change with the test condition

Table 3-1: Core Power Table

Table number	Time [s]	Fraction of Test Nominal Power
20290001	0.	1.0
20290002	5.25	0.7561302
20290003	10.4	0.3840996
20290004	15.6	0.2160919
20290005	26.04	0.1088122
20290006	31.26	0.0881226
20290007	72.8	0.0459770
20290008	166.5	0.0354406
20290009	331.5	0.0266283
20290010	762.5	0.0208620
20290011	1459.5	0.0178160
20290012	4907.5	0.0122605

3.2.7 Intact Loop Steam Generator Vessel Top Heat Losses

Table 3-10 shows the HTC value for the IL SG top part of the vessel.

Table 3-10: Intact Loop Steam Generator Vessel Top HTC table

Table number	Time [s]	HTC [W/m²/K]
20265001	0.	0.01
20265002	10000.	0.01

3.2.8 Broken Loop Steam Generator Vessel Bottom Heat Losses

Table 3-11 shows the HTC value for the BL SG bottom part of the vessel

Table 3-11: Broken loop Steam generator Vessel Bottom HTC table

Table number	Time [s]	HTC [W/m²/K]
20270001	0.	6.0
20270002	10000.	6.0

3.2.9 Broken Loop Steam Generator Vessel Top Heat Losses

Table 3-12 shows the HTC value for the BL SG top part of the vessel

Table 3-12: Broken Loop Steam Generator Vessel Top HTC table

Table number	Time [s]	HTC [W/m²/K]
20275001	0.	0.01
20275002	10000.	0.01

4 MATERIAL PROPERTY

This section of the engineering handbook documents the material property in the input deck. The stored material property in the RELAP5 codes have not been used. The option TBL/FCTN in card 201MMM00 has been used for the different materials, the thermal conductivity and the heat capacity are input as a function of temperature.

4.1 Inconel 625

Table 4-1 list the input entries for the inconel 625 heat conductivity and *Table 4-2* the heat capacity property.

Table 4-1: Inconel 625 Heat Conductivity

Table number	Temperature [K]	Heat conductivity [W/(m K)]
20100101	93.	12.
20100102	473.	12.5
20100103	573.	13.9
20100104	673.	15.3
20100105	2073.	16.3

Table 4-2: Inconel 625 Heat Capacity

Table number	Temperature [K]	Heat Capacity [J/(kg K)]
20100151	93.	$3.46\cdot10^{+6}$
20100152	373.	$3.67\cdot10^{+6}$
20100153	473.	$3.87\cdot10^{+6}$
20100154	573.	$4.05\cdot10^{+6}$
20100155	673.	$4.26\cdot10^{+6}$
20100156	2073.	$4.36\cdot10^{+6}$

ENGINEERING HANDBOOK

Structure & Samples

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5 LOGIC AND CONTROL SYSTEM

5.1 Control Variables

5.1.1 Level

Table 5-1 summarized the level control variable that are present in the present *Table 5-1*, the control variable related to a particular part of the ITF (the same approach has been used in the input file). For each control variable t measurement is identified (second column), the correspondence with the ITF m given in the fifth column. For each control variable the last column of *Table 5-1* to the section that the described the specific control variable.

5.1.1.1 Pressurizer level

The pressurizer level is calculated summing the liquid void fraction in each elevation change of each cell for which the variable *"voidf"* is calculated (*Equa 2-2*). Two control variable are used to calculated the actual collapsed level of and 4309.

Control variable 4209:

$$PRZ\_L1 = 0.395 \cdot voidf\ 420\_01 + 0.395 \cdot voidf\ 425\_01 + 0.585 \cdot voidf\ 430\_01 + 0.5 \cdot voidf\ 430\_02 + 0.5 \cdot voidf\ 430\_03 + 0.5 \cdot voidf\ 430\_04 + 0.5 \cdot voidf\ 430\_05 + 0.345 \cdot voidf\ 430\_06$$

Control variable 4309:

$$PRZ\_L = 0.336 \cdot voidf\ 430\_07 + 0.5 \cdot voidf\ 430\_08 + 0.5 \cdot voidf\ 430\_09 + 0.6 \cdot voidf\ 430\_10 + 0.6 \cdot voidf\ 430\_11 + 0.705 \cdot voidf\ 430\_12 + 0.705\ voidf\ 440\_01 + 1.0 \cdot cntrlvar\ 4209$$

5.1.1.1 Intact Loop Level

Intact Loop Steam Generator Inlet Global Level (CNTRLVAR 1159)

Control variable 1159 calculates the collapsed liquid level in the inlet pipe of generator, it corresponds to measurement channel "CL90AB" +1.19 m – 0.055 level is calculated summing the liquid void fraction in each cells multiplied by t each cell for which the variable *"voidf"* is calculated (*Equation 5-3*). The initial v 2.9 m.

Control variable 1159:

$$ILSGIN\_L = 0.83 + 0.458 \cdot voidf\ 110\_08 + 0.4 \cdot voidf\ 110\_09 + 0.4 \cdot voidf\ 110\_10 + 0.3 \cdot voidf\ 110\_11 + 0.2 \cdot voidf\ 110\_12 + 0.312 \cdot voidf\ 115\_01$$

General Location	Location	Control Variable Number	Control variable Name	Experimental Channel Measurement Correspondence	Description	Reference
IL LEVEL	Pressurizer	4209	PRZ_L1	-		5.1.1.1
		4309	PRZ_L	-		
	IL SG inlet global	1159	ILSGIN-L	CL90AB+1.19m-0.055m		5.1.1.1
	IL U-tubes ascending side	1189	ILUTAS-L	CL90BP+2.995m		
		1199	ILUTAS-L			
	IL U-tubes descending side	1219	ILUTDS-L	-		
		1229	ILUTDS-L	CL92BP+2.955m		
	IL SG outlet Global level	1259	ILSGOT-L	- CL93AB+1.19m-0.055m		
	IL Loop Seal	1299	ILLS-1	-		
		1309	ILLS-2	-		
		1319	ILLS-L	CL1792X3		
BL LEVEL	BL SG Inlet global	2159	BLSGIN-L	CL80+1.045m -0.02m		5.1.1.2
	BL U-tubes ascending side	2189	BLUTAS-L	-		
		2199	BLUTAS-L	CL80BP+2.95m		
	IL U-tubes descending side	2219	ILUTDS-L	-		
		2229	ILUTDS-L	CL82BP+2.95m		
	BL SG outlet Global	2259	BLSGOT-L	CL82AB+0.045m-0.02m		
	BL Loop Seal	2309	BLLS-1			
		2319	BLLS-L	CL2782x2		
RPV LEVEL	RPV Core Level	3295	RPVCOR-1	-		5.1.1.3
		3309	RPVCOR-L	-		
	RPV Riser Level	3159	RPVRSR-1	-		
		3409	RPVRSR-3	-		
		3459	RPVRSR-L	CL3RYA	Approximately	
	RPV Downcomer Level	3009	RPVDC-1	-		
		3019	RPVDC-2	-		
		3029	RPVDC-3	-		
		3059	RPVDC-L	CL3DYB+0.17m		
IL SG	IL SG Downcomer level	6009	ILSGDC-1	-		5.1.1.4
		6359	ILSGDC-L	-		
BL SG	BL SG Downcomer level	7009	BLSGDC-1	-		5.1.1.4
		7359	BLSGDC-L	-		
IL SG	IL SG Riser level	6049	ILSGRS-1	-		5.1.1.4
		6059	ILSGRS-L	-		
BLSG	BL SG Riser level	7049	BLSGRS-1	-		5.1.1.4
		7059	BLSGRS-L	-		

5.1.5 Pressure Drop

Table 5-5: Summary Table for Pressure Drop Control Variable summarized the control variables set up in the RELAP5 input to calculate the pressure drop. In the present Table 5-5: Summary Table for Pressure Drop Control Variable, the control variable related to a particular part of the ITF are grouped together (the same approach has been used in the input file). For each control variable the identification of the corresponding pressure drop measurement is identified (fifth column) For each control variable the last column of *Table 5-5* provide the reference to the section that the described the specific control variable. The pressure drop are calculated as a pressure difference between two specific cells.

Table 5-5: Summary Table for Pressure Drop Control Variable

General Location	Location	Control Variable Number	Control variable Name	Experimental Channel Measurement Correspondence	Description	Reference
RPV	DC-RSR	3443	PV.DC-RS	PD3D3RBA		5.1.5.1
	DC	3003	PVDC-PD	PD3DBT		
	DC-RSR	3153	PVCOR-PD	PD3D3RUU		
	RSR	3303	PVCOR-PD	PD3RUG		
		3403	PVUP-PD	PD3RGA		
		3453	PVRSR-PD	PD3RYA		
	UH-UP	3703	UPUH-PD	PD3R29A		
	UP-HL	1003	UP-IL.HL	PD3R11A4		
IL	IL-CL	1903	IL.CL-HL	PD161133		5.1.5.2
	IL-HL	1103	ILHL-PD	PD1190A		
	IL-UT ascending	1213	ILUTA-PD	PD90BPX2		
	IL-UT	1203	ILUT-PD	PD9092AA		
	Loop seal descending	1253	ILUT-PD	PD9217A		
	Loop seal ascending	1303	ILLSA-PD	PD1714		
	MCP	1403	ILMCP-PD	PD1151456		
	CL-DC	1703	IL.CL-DC	PD163DB3		
BL	UP-HL	2003	UP-BL.HL	PD3R21A4		5.1.5.3
	CL-HL	2903	BL.CL-HL	PD262133		
	HL	2103	BLHL-PD	PD21180A		
	BL-UT ascending	2213	BLUTA-PD	PD80BPX2		
	BL-UT	2203	BLUT-PD	PD8082AA		
	Loop seal descending	2253	BLLSD-PD	PD8227A		
	Loop seal ascending	2303	BLLSA-PD	PD2724		
	MCP	2403	BLMCP-PD	PD252451		
	CL-DC	2703	BL.CL-DC	PD263DB7		

5.1.5.1 Reactor Pressure Vessel

Downcomer-Riser Pressure Drop (CNTRLVAR 3443)

The pressure drop is calculated as a pressure difference between cell 30001 and cell 34501, the resulting value compared to the measured pressure drop identified by the abbreviation "PD3D3RBA".

$$PV.DC - RS = 1.0 \cdot p\,30001 + (-1.0) \cdot p\,34501 \qquad \text{Equation 5-69}$$

5.3 Logical and Variable Trips

Table 5-6 summarized the primary and secondary side trips used in the input deck that are detailed in the present section. Each trip should be adjust according to the specific set point for each different test.

Table 5-6: Summary Table for Primary and Secondary Side Trips.

Trip Number	Trip Type	Trip Function	Component Controlled	Reference
Primary Side Trip				
251	Variable	Control the actuation of the BL pump locked rotor resistance simulator	HC 251	5.3.1.1
252	Variable	Main Coolant Pump brake	VT 253	
253	Variable	Rotor simulator valve closure initiation	LT (1)251	
254	Variable	Rotor simulator valve closure end	LT (1)251	
(1)251	Logical	BL pump locked rotor valve closure trip	HC 251	
388	Variable	UH-DC valve close trip	HC 388	5.3.1.2
389	Variable	UH-DC valve open trip		
420	Variable	PRZ heaters power-off	CS 4358	5.3.1.3
181	Variable	IL seal water table trip	HC 181	5.3.1.4
281	Variable	BL seal water table trip	HC281	
398	Variable	Pumps seal water drain close	LT (1)398	
(1)398	Logical	Pumps seal water drain table	HC 398	
345	Variable	Upper Plenum Lower Pressure Signal	-	5.3.1.5
(1)900	Logical	Core power trip	T 900	5.3.1.6
140	Variable	MCP IL trip	HC 140	5.3.1.7
141	Variable	MCP IL decay velocity activated		
240	Variable	MCP BL trip	HC240	
241	Variable	MCP BL decay velocity activated		
810	Variable	IL ACC CL LINE valve open trip	HC 810	5.3.1.8
811	Variable	IL ACC CL LINE trip		
820	Variable	IL ACC HL LINE valve open trip	HC 820	
821	Variable	IL ACC HL LINE trip		
910	Variable	BL ACC CL LINE valve open trip	HC 910	
911	Variable	BL ACC CL LINE trip		
920	Variable	BL ACC HL LINE valve open trip	HC 920	
921	Variable	BL ACC HL LINE trip		
850	Variable	HPIS Signal	VT 855	5.3.1.9
855	Variable	HPIS delay	HC 855	
90	Variable	Break open	HC 090	5.3.1.10
91	Variable	Break close		

(to be continued)

CONCLUSIONS

- A procedures for a creation of a qualified experimental database has been developed and adopted
- Review of each document is intrinsic in the procedure

- RDS collects the most important geometrical data of the facility and gives calculated values directly usable from the input developers
- RDS is a powerful document that follows the IAEA guidelines
- QR assures that the calculated value fulfill pre determined acceptability criteria.
- EH provides engineering justification of the input deck entries
- EH links the RDS of the facility, the code and the R5-3D input deck

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GRUPPO DI RICERCA NUCLEARE – SAN PIERO A GRADO (GRNSPG)

OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluation

***Barcelona, Spain,* 16-18 November 2011**

A Procedure for Characterizing the Range of Input Uncertainty Parameters by the Use of FFTBM

A. Kovtonyuk , A.Petruzzi, M. Raucci, D. De Luca, F. Veronese, and F.D'Auria

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Contents

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- FFTBM Details
- FFTBM to characterizing IP and range of IP
  - Method
  - Investigated Criteria
  - Preliminary applications
    - Marviken Test
    - Edward Pipe
    - LOBI Test A1-83 (10% LOCA)

FFTBM Details

Generally, the starting point of each method to quantify the accuracy is an error function, ΔF. Some requirements were fixed which an objective error function ΔF should satisfy:

1) AT ANY TIME OF THE TRANSIENT THIS FUNCTION SHOULD REMEMBER THE PREVIOUS HISTORY;
2) ENGINEERING JUDGMENT SHOULD BE AVOIDED OR REDUCED;
3) THE MATHEMATICAL FORMULATION SHOULD BE SIMPLE;
4) THE FUNCTION SHOULD BE NON-DIMENSIONAL;
5) IT SHOULD BE INDEPENDENT UPON THE TRANSIENT DURATION;
6) COMPENSATING ERRORS SHOULD BE TAKEN INTO ACCOUNT (OR POINTED OUT);
7) ITS VALUES SHOULD BE NORMALIZED.

FFTBM Details

Possible Solutions for Accuracy Quantification

WHEN TWO CORRESPONDING SETS OF DATA, OR EVEN TWO VALUES, ARE AVAILABLE (A MEASURED AND A CALCULATED VALUE), AN INFINITE NUMBER OF PARAMETERS CAN BE USED TO MARK THE DIFFERENCE.

IN THE CASE OF SYSTEM THERMAL-HYDRAULICS, FOR ANY QUANTITY THE '**QUANTITY VALUE**' AND '**THE TIME WHEN THE VALUE OCCURS**' CAN BE DISTINGUISHED.

TIME INTEGRALS CAN BE PERFORMED FOR TIME DEPENDENT QUANTITIES, AS WELL AS FOR THE DIFFERENCE BETWEEN MEASURED AND CALCULATED VALUES.

SQUARE OF THE DIFFERENCES CAN BE CONSIDERED, TOO.

FFTBM Details

Possible Solutions for accuracy Quantification

NO-ONE OF THESE APPROACHES REVEALS FULLY SATISFACTORY, IF JUDGMENT IS NECESSARY FOR DIFFERENT TRANSIENT TYPES (E.G. TRANSIENT LASTING 50 s OR 50000 s).

THE NEED TO APPLY ANY CODE TO DIFFERENT DURATION TRANSIENTS AND TO ESTABLISH A COMMON BASIS FOR THE EVALUATION OF THE CALCULATION PERFORMANCES, SUGGESTED THE EXPLOITATION OF THE **FREQUENCY DOMAIN**.

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FFTBM Details

FAST FOURIER TRANSFORM

A fundamental property of the Fourier Transform (FT) consists in the capability to analyze in the frequency domain any relationship between two quantities taken from the time domain without loss of information.

The Fourier transform can translate a given time function g(t), in a corresponding complex function defined, in the frequency domain, by the relationship:

$$\tilde{g}(f) = \int_{-\infty}^{+\infty} g(t) \cdot e^{-j \cdot 2\pi \cdot f \cdot t} dt$$

Experimental and Calculated trends shall verify the analytical conditions required by its application theory:

- it is assumed that they are continuous (or generally continuous) in the considered time intervals with their first derivatives
- and absolutely integrable in the interval (- ∞, + ∞). This last requirement can be easily satisfied in our case, since the addressed functions assume values different from zero only in the interval (0, T). Therefore:

$$\tilde{g}(f) = \int_{0}^{T} g(t) \cdot e^{-j \cdot 2\pi \cdot f \cdot t} dt$$

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FFTBM Details

The Fourier integral is not suitable for machine computation (infinity of samples of g(t) is required). Thus, it is necessary to truncate the sampled function g(t) (only a finite number of points have to be considered) or in other words, the discrete Fourier transform is evaluated.

When using functions sampled in digital form, the FFT (Fast FT) can be used, i.e. algorithm that computes more rapidly the discrete Fourier Transform. In order to apply this algorithm, functions must be identified by a number of points, 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 frequencies fn = n/T, (n = 0, 1...2m), in which T is the time duration of the sampled signal.

The accuracy quantification of a code calculation considers the amplitude, in the frequency domain, of the experimental signal Fexp(t) and the error function:

$$\Delta F = F_{calc}(t) - F_{exp}(t) \qquad (1)$$

FFTBM Details

The method characterizes each calculated and corresponding measured quantity through the dimensionless average amplitude (AA) and the weighted frequency (WF):

$$AA = \frac{\sum_{n=0}^{2^m} \left|\widetilde{\Delta F}(f_n)\right|}{\sum_{n=0}^{2^m} \left|\widetilde{F}_{exp}(f_n)\right|} \qquad (2)$$

$$WF = \frac{\sum_{n=0}^{2^m} \left|\widetilde{\Delta F}(f_n)\right| \cdot f_n}{\sum_{n=0}^{2^m} \left|\widetilde{F}_{exp}(f_n)\right|} \qquad (3)$$

FFTBM Details

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% error).

The **WF** factor characterizes the **kind of error**, because its value emphasizes if the error has more relevance at low or high frequencies. Depending upon the transient, high frequency errors can be more acceptable than low frequency ones. In other terms, *better accuracy is achieved by low AA values at high WF values.*

Trying to give an overall picture of the accuracy of a given calculation, average indexes of performance are obtained by defining:

$$(AA)_{tot} = \sum_{i=1}^{N_{var}} (AA)_i \cdot (wf)_i \qquad (4)$$

$$(WF)_{tot} = \sum_{i=1}^{N_{var}} (WF)_i \cdot (wf)_i \qquad (5)$$

FFTBM Details

where:

$$\sum_{i=1}^{N_{var}} (wf)_i = 1 \qquad \textbf{(6)}$$

Nvar is the number of the analyzed parameters and (wf)i are weighting factors introduced to take into account the different importance of each parameter from the viewpoint of safety analyses.

Each (*wf*)i takes into account:

- **experimental accuracy**: experimental trends of thermal-hydraulic parameters are characterized by a more or less sensible uncertainty due to intrinsic characteristic of instruments, method of measure, adopted way to compare experimental measures and the code calculated results;
- **safety relevance**: 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.

FFTBM Details

Further contribution is given by a factor that normalizes the AA value calculated for the selected parameters with respects to the AA value calculated for the primary pressure. This factor has been introduced in order to consider the physic relations existing between different quantities (i.e. fluid temperature and pressure in case of saturated blow-down must be characterized by the same order of error).

The weighting factor of the j-th parameter is defined as:

$$(wf)_j = \frac{(W_{exp})_j \cdot (W_{saf})_j \cdot (W_{norm})_j}{\sum_{j=1}^{N_{var}} (W_{exp})_j \cdot (W_{saf})_j \cdot (W_{norm})_j} \qquad \textbf{(7)}$$

wexp is the contribution related to the experimental accuracy

wsaf is the contribution which expresses the safety relevance of the addressed parameter

wnorm is the component of normalization with reference to the average amplitude evaluated for the primary side pressure

FFTBM Details

This introduces a degree of engineering judgment that has been fixed by a proper and unique definition of the weighting factors:

Parameter	W_{exp}	W_{saf}	W_{norm}
Primary pressure	1.0	1.0	1.0
Secondary pressure	1.0	0.6	1.1
Pressure drops	0.7	0.7	0.5
Mass inventories	0.8	0.9	0.9
Flow rates	0.5	0.8	0.5
Fluid temperatures	0.8	0.8	2.4
Clad temperatures	0.9	1.0	1.2
Collapsed levels	0.8	0.9	0.6
Core power	0.8	0.8	0.5

Tab. IV - Selected weighting factor components for typical thermalhydraulic parameters

FFTBM Details

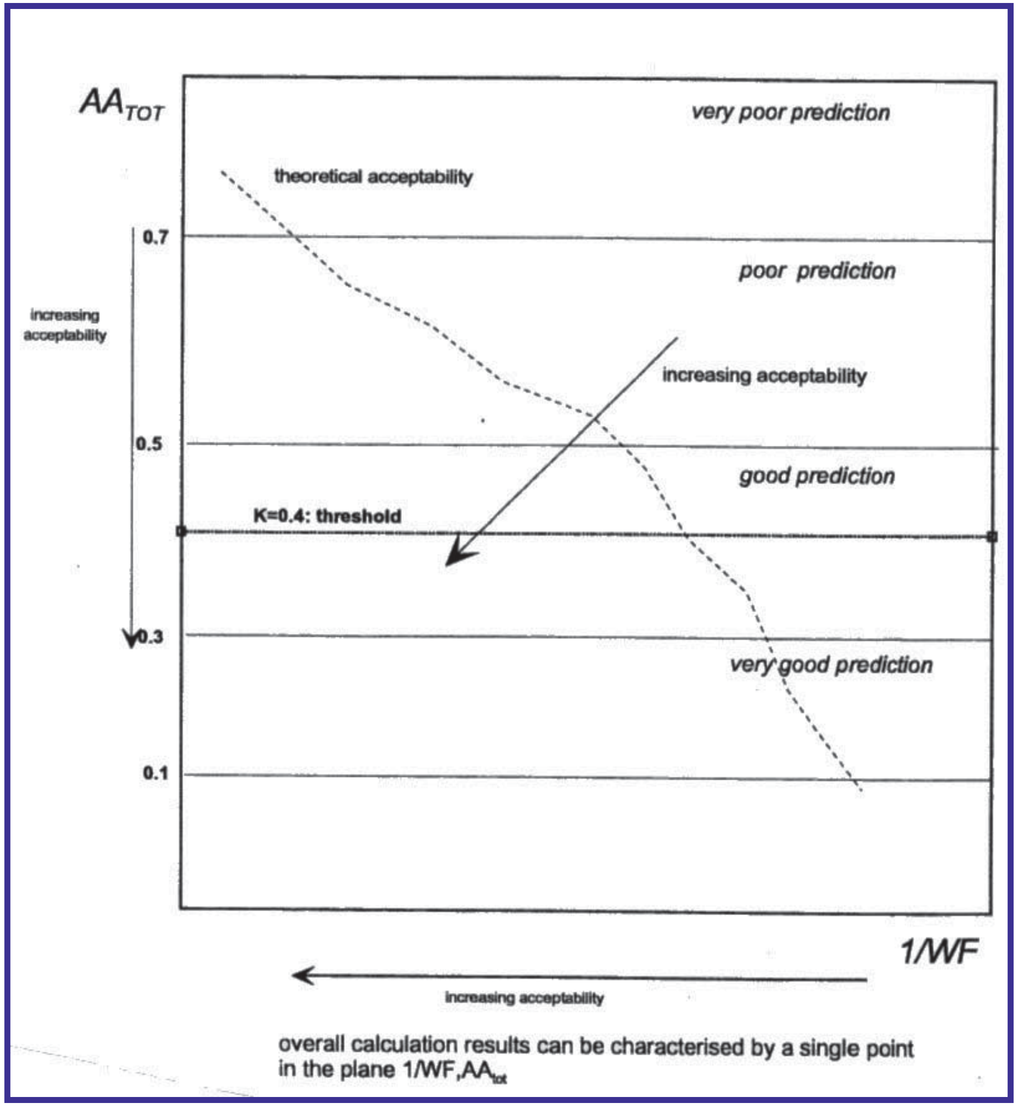
The most suitable factor for the definition of an acceptability criterion, therefore for using the method, is the average amplitude AA. With reference to the accuracy of a given calculation, we can define the following acceptability criterion:

$$\boxed{(AA)_{tot} < K} \qquad (8)$$

where K is an acceptability factor valid for the whole transient. As lower is the $(AA)_{tot}$ value, as better is the accuracy of the analyzed calculation (i.e. the code prediction capability and acceptability is higher).

$(AA)_{tot}$ should not exceed 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.

FFTBM Details



- **'very poor/unacceptable'**

  $(AA)_{tot} > 0.7$

- **'poor'**

  $0.5 < (AA)_{tot} \leq 0.7$

- **'good'**

  $0.3 < (AA)_{tot} \leq 0.5$

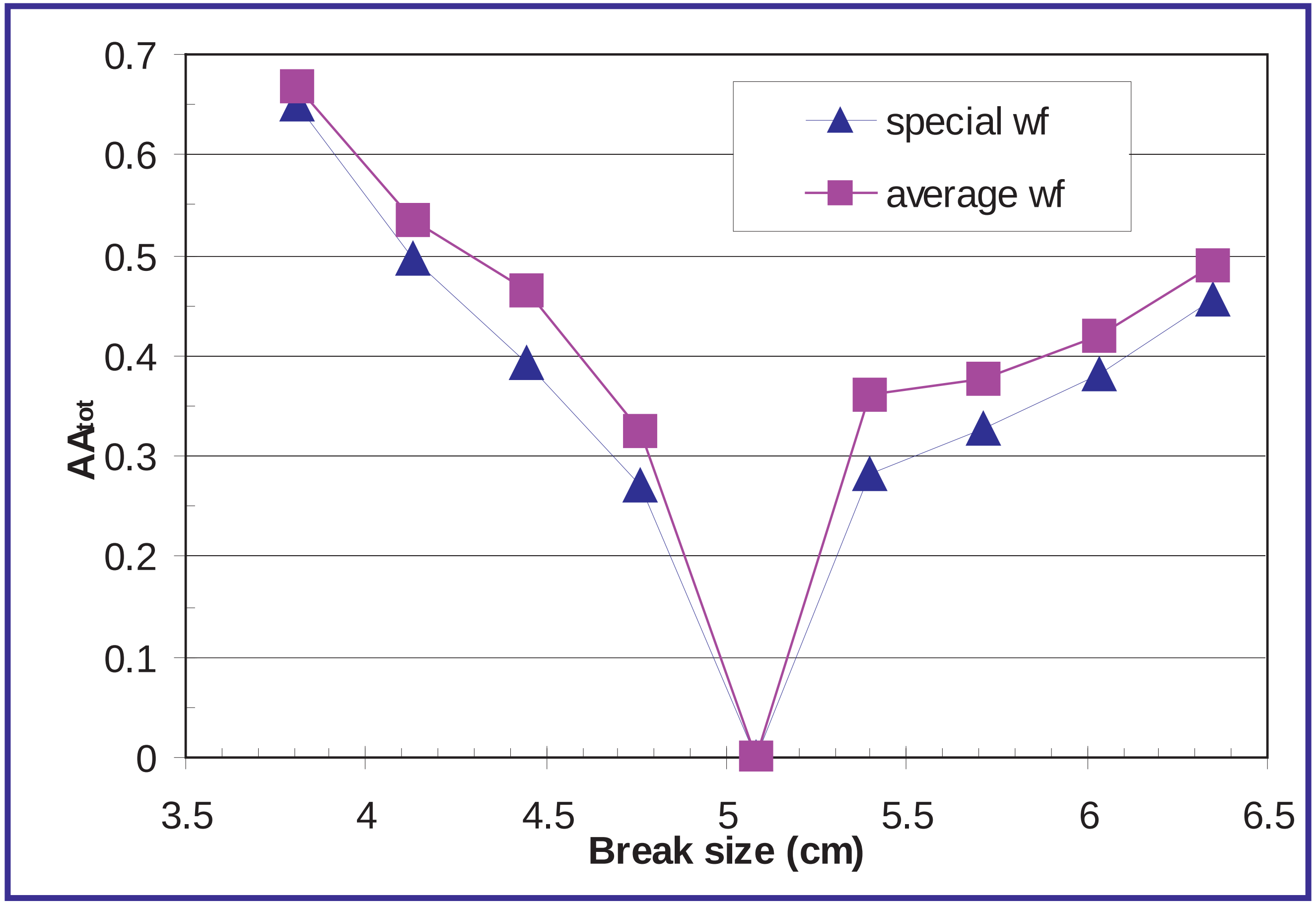
- **'very good'**

  $(AA)_{tot} \leq 0.3$

FFTBM Details

(from the paper by Prosek et al.)

TESTING THE VALIDITY OF THE FFTBM THROUGH THE EVALUATION OF NPP SBLOCA SCENARIO



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FFTBM Details

- **PIONEERING WORK PERFORMED BY OECD/CSNI TASK GROUP ON THERMALHYDRAULICS IN THE YEARS 1985-89**

- **PROPOSAL FOR A METHOD FOR ACCURACY QUANTIFICATION (*)**

  Ambrosini W., Bovalini R., D'Auria F. "Evaluation of Accuracy of Thermalhydraulic Codes Calculations“ J. Energia Nucleare, Vol. 7 N. 2, May 1990

- **QUALITATIVE AND QUANTITATIVE ACCURACY EVALUATION**

  D'Auria F., Galassi G.M. "Code Validation and Uncertainties in System Thermalhydraulics“ J. Progress in Nuclear Energy, Vol 33 No 1/2, pp 175-216, 1998

- **OVERVIEW OF METHODS FOR QUANTITATIVE ACCURACY EVALUATION**

  Kunz R.F., Kasmala G.F., Mahaffy J.H., Murray C.J "On the Automated Assessment of Nuclear reactor systems code accuracy“ J. Nuclear Engineering and Design, Vol 211, Nos 2 and 3 (2002)

(*) FFTBM discussed hereafter, utilized by different Institutions

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FFTBM to characterizing IP and range of IP

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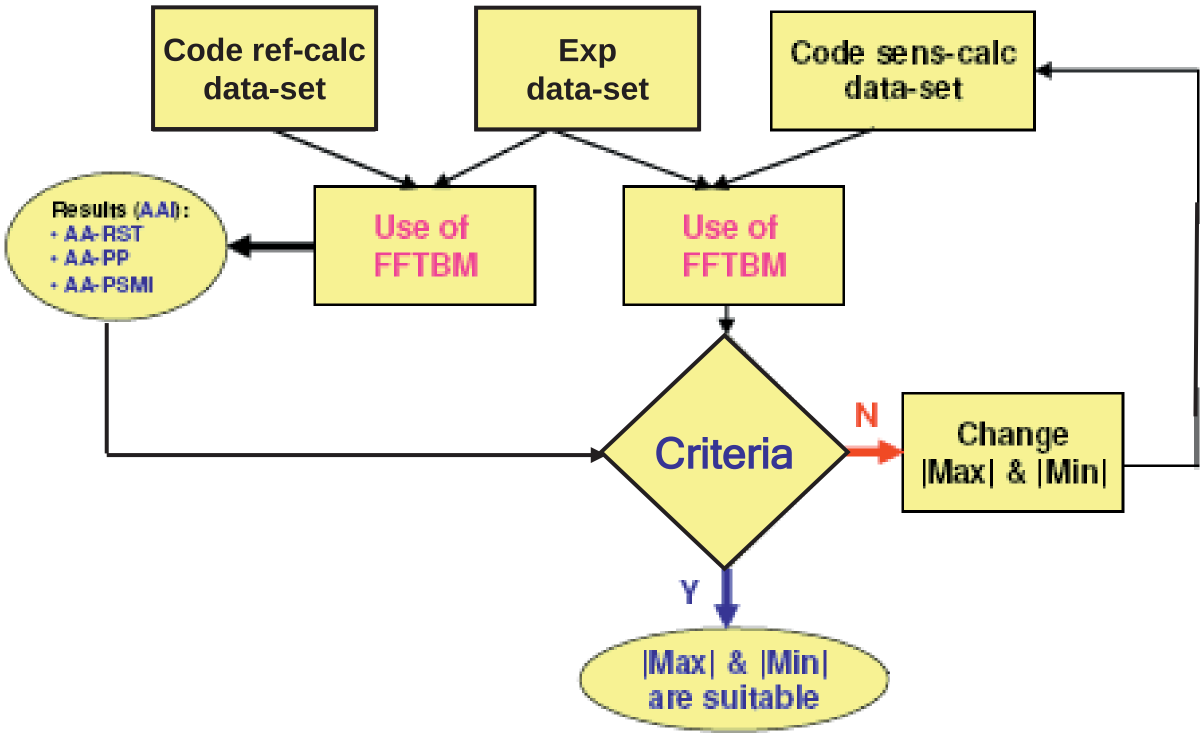

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Objective of Specifications for PREMIUM Phase II

Method

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- ❑ Running Reference Case (RC)
- ❑ Selection of Responses
- ❑ Derivation by FFTBM of AA^{REF} for each selected response
- ❑ To define CRiteria (CR) for deriving the range of input parameters (part of development process of the method)
- ❑ To select a set of Input Uncertainty Parameters
- ❑ To run Sensitivity cases and perform a qualitative check
- ❑ To apply FFTBM to the sensitivity cases AA*
- ❑ To apply CR for identifying the Range
- ❑ To discard not relevant Input Uncertainty Parameters

Objective of Specifications for PREMIUM Phase II

Investigated Criteria

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1. CRITERIUM CR1

CR1.a $AA_p^{(*)} < 0.1$

CR1.b $MAX(AA_{mf}^{(*)}/AA_{mf}^{(ref)};AA_{Td}^{(*)}/AA_{Td}^{(ref)};AA_{Tu}^{(*)}/AA_{Tu}^{(ref)})-1 < 0.$

2. CRITERIUM CR2

CR2.a $AA_p^{(*)} < 0.1$

CR2.b
$$\begin{cases} AA_G^{(*)} := \sqrt{AA_p^{(*)\,2} + AA_{mf}^{(*)\,2} + AA_{Td}^{(*)\,2} + AA_{Td}^{(*)\,2}} \\ AA_G^{(*)}/AA_G^{(ref)} - 1 < P1 \end{cases}$$

CR2.c $MAX(AA_{mf}^{(*)}/AA_{mf}^{(ref)};AA_{TD}^{(*)}/AA_{TD}^{(ref)};AA_{TU}^{(*)}/AA_{TU}^{(ref)}) - 1 < P2$

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Objective of Specifications for PREMIUM Phase II

Investigated Criteria

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3. CRITERIUM CR3

CR3.a $AA_p^{(*)} < 0.1$

CR3.b
$$\begin{cases} AA_G^{(*)} := \sqrt[4]{AA_p^{(*)} \cdot AA_{mf}^{(*)} \cdot AA_{Td}^{(*)} \cdot AA_{Td}^{(*)}} \\ AA_G^{(*)} / AA_G^{(ref)} - 1 < P1 \end{cases}$$

CR3.c $\mathrm{MAX}(AA_{mf}^{(*)} / AA_{mf}^{(ref)}; AA_{TD}^{(*)} / AA_{TD}^{(ref)}; AA_{TU}^{(*)} / AA_{TU}^{(ref)}) - 1 < P2$

4. CRITERIUM CR4

CR4.a $AA_p^{(*)} < 0.1$

CR4.b
$$\begin{cases} AA_G^{(*)} := \sqrt{\left(\frac{AA_p^{(*)}}{AA_p^{(ref)}}\right)^2 + \left(\frac{AA_{mf}^{(*)}}{AA_{mf}^{(ref)}}\right)^2 + \left(\frac{AA_{Td}^{(*)}}{AA_{Td}^{(ref)}}\right)^2 + \left(\frac{AA_{Tu}^{(*)}}{AA_{Tu}^{(ref)}}\right)^2} \\ AA_G^{(*)} / AA_G^{(ref)} - 1 < P1 \end{cases}$$

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Objective of Specifications for PREMIUM Phase II

Investigated Criteria

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5. CRITERION CR5

CR5.a $AA_p^{(*)} < 0.1$

CR5.b
$$\begin{cases} AA_G^{(*)} := \sqrt{\dfrac{\sum_i \left(AA_i^{(*)}\right)^2}{\sum_i \left(AA_i^{(ref)}\right)^2}} \\ AA_G^{(*)} / AA_G^{(ref)} - 1 < P1 \end{cases}$$

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Objective of Specifications for PREMIUM Phase II

Investigated Criteria

6. CRITERION CR6

CR6.a $AA_p^{(*)} < 0.1$

CR6.b
$$\begin{cases} AA_G^{*,IP} / AA_G^{REF} - 1 < T1 \\ AA_G^{*,IP} = \dfrac{1}{N} \cdot \sum_{i=1}^{N} AA_{R_i}^{*,IP} \end{cases}$$

7. CRITERION CR7

CR7.a $AA_p^{*,IP} < 0.1$

CR6.b
$$\begin{cases} AA_G^{*,IP} - 1 < T1 \\ AA_G^{*,IP} = \sqrt{\dfrac{\sum_{i=1}^{N} \left(AA_{R_i}^{*,IP} \cdot w_{f_i}\right)^2}{\sum_{i=1}^{N} \left(AA_{R_i}^{REF} \cdot w_{f_i}\right)^2}} \end{cases}$$

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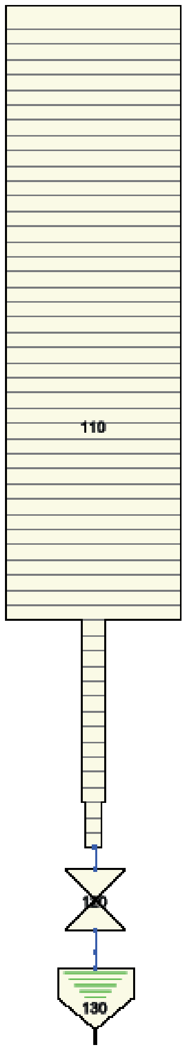
Objective of Specifications for PREMIUM Phase II

Preliminary applications: Marviken CFT04



- **RELAP5\M3.3 p03**
  - Vessel body (40 volumes)
  - Discharge pipe (12 volumes)
  - Discharge Nozzle (3 volumes) – to be varied depending on test
- **Selected responses:**
  - Pressure (P)
  - Break Flow Rate (MF)
  - Fluid Temperature @ top (TU)
  - Fluid temperature @ bottom (TD)
- **Set of Input Parameters (about 20)**

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Objective of Specifications for PREMIUM Phase II

Preliminary applications: Marviken CFT04

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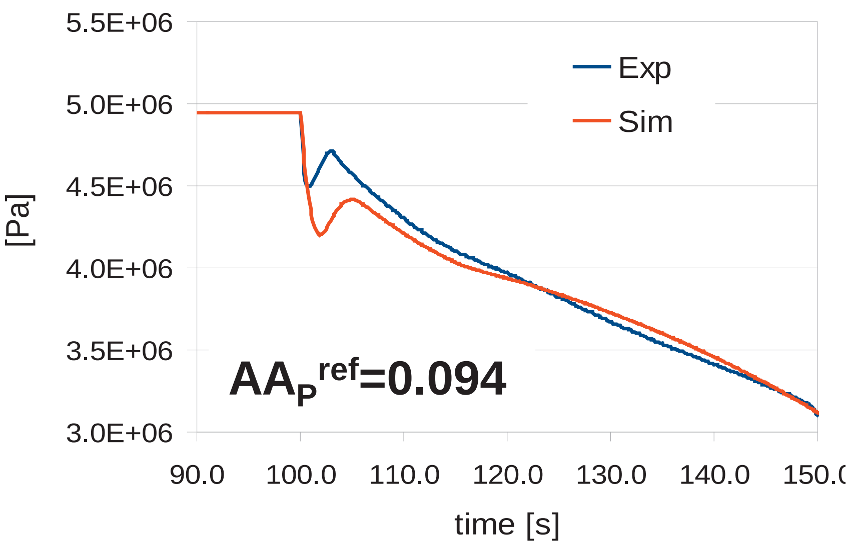

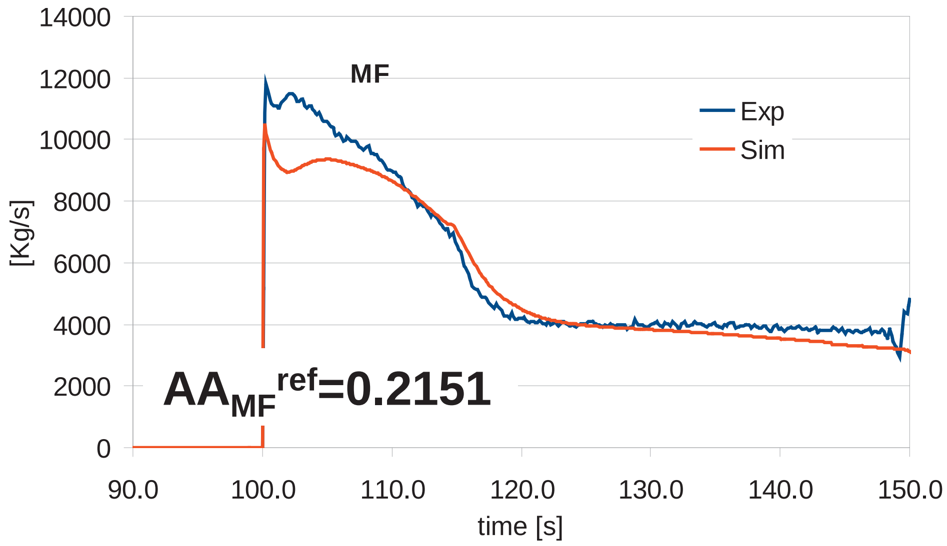

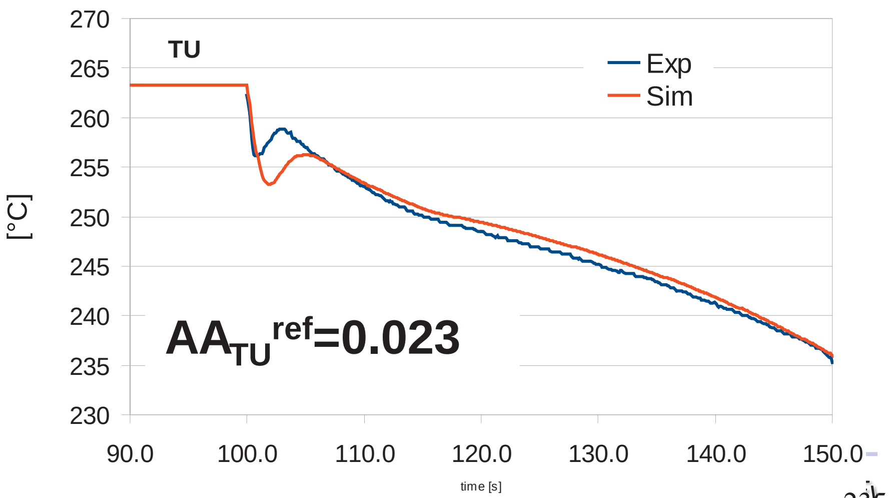

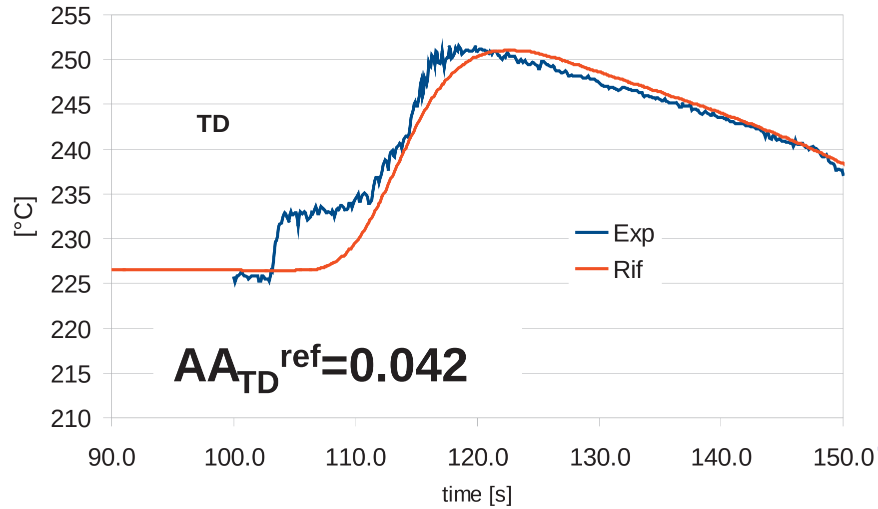

Objective of Specifications for PREMIUM Phase II

Preliminary applications: Marviken CFT04

Selection of Input Parameter

1. "Henry-Fauske" choked flow model, discharge coefficient (RC = 0.8)
2. "Henry-Fauske" choked flow model, Thermal Non Equlibrium Constant (RC = 0.14)
3. Initial water level in the vessel (RC = 0.4 m)
4. Temperature difference across the transition zone
5. Upper-dome pressure
6. Elevation of the Transition zone
7. Fictitious K-loss value

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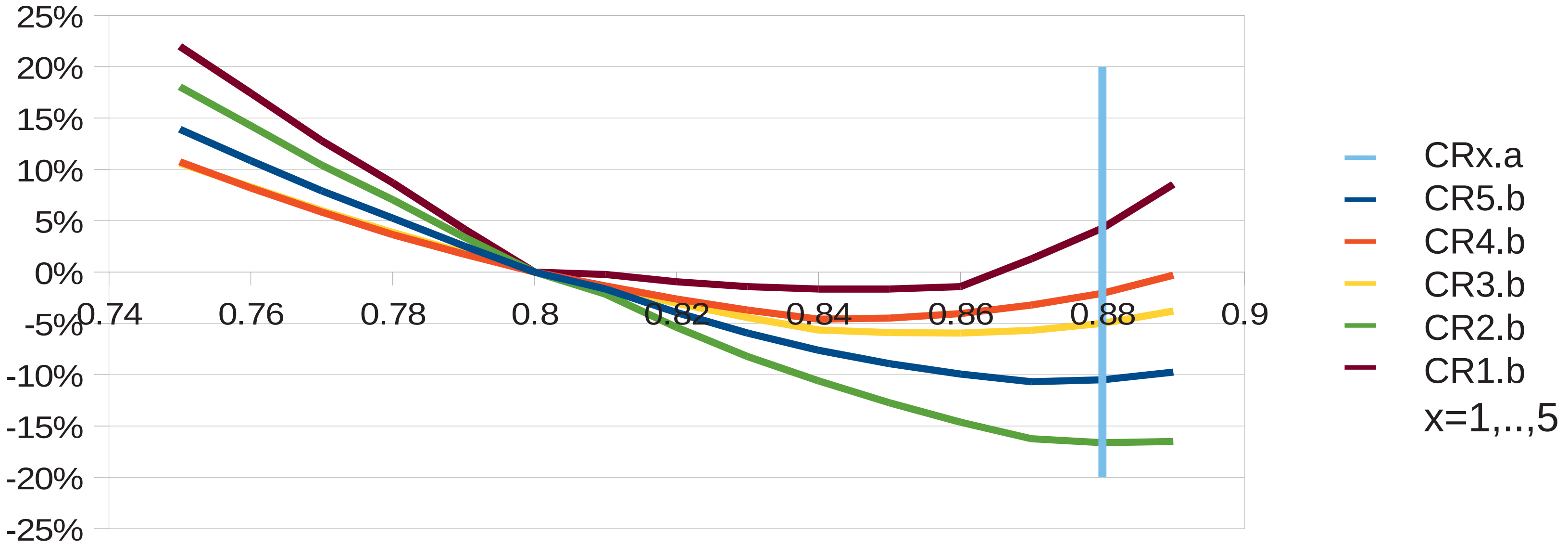
Objective of Specifications for PREMIUM Phase II

Preliminary applications: Marviken CFT04

1. "Henry-Fauske" choked flow model, discharge coefficient (RC = 0.8)



CR1: CR1.a '+' CR1.b

CR2: CR2.a '+' CR2.b(P1=0) '+' CR2.c(P2=0) [CR2.c(P2=0) = CR1.b]

CR3: CR3.a '+' CR3.b(P1=0) '+' CR3.c(P2=0) [CR3.c(P2=0) = CR1.b]

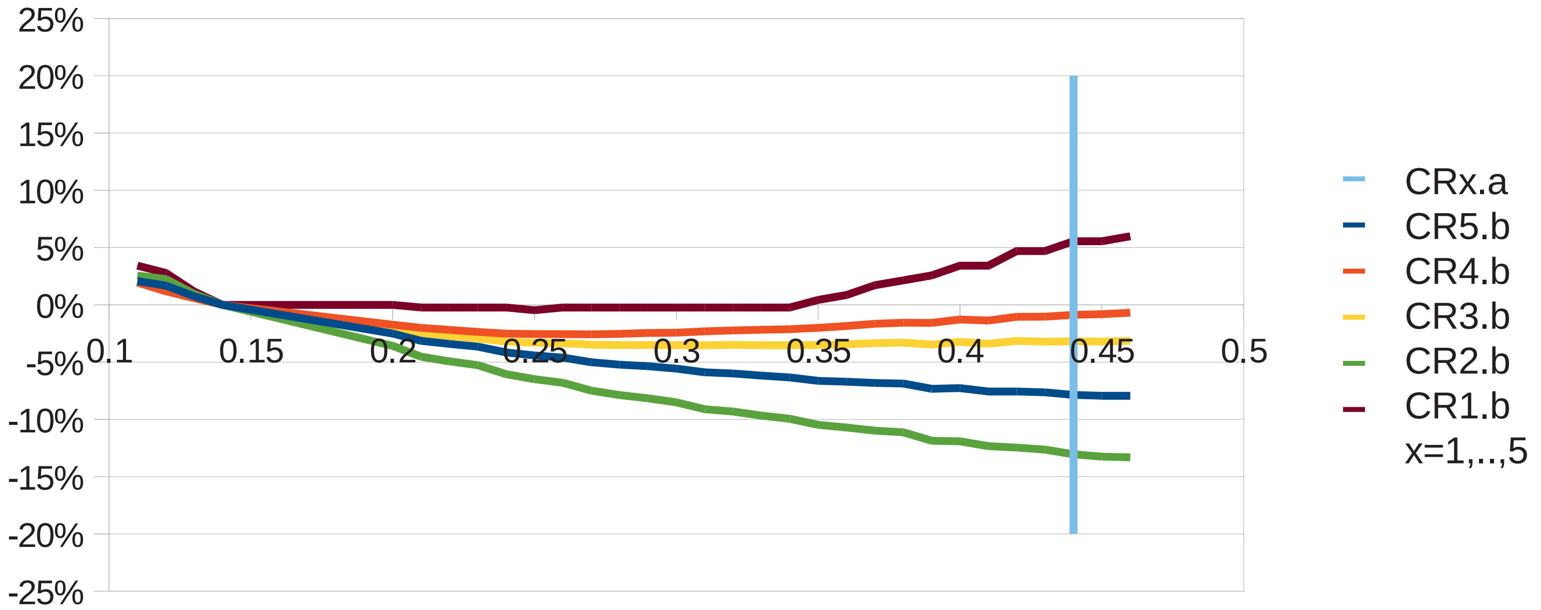
CR4: CR4.a '+' CR4.b(P1=0)

CR5: CR4.a '+' CR5.b(P1=0)

Objective of Specifications for PREMIUM Phase II

Preliminary applications: Marviken CFT04

2. Henry-Fauske" choked flow model, Thermal Non Equilibrium Constant (RC = 0.14)



CR1: CR1.a '+' CR1.b

CR2: CR2.a '+' CR2.b(P1=0) '+' CR2.c(P2=0) [CR2.c(P2=0) = CR1.b]

CR3: CR3.a '+' CR3.b(P1=0) '+' CR3.c(P2=0) [CR3.c(P2=0) = CR1.b]

CR4: CR4.a '+' CR4.b(P1=0)

CR5: CR4.a '+' CR5.b(P1=0)

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Objective of Specifications for PREMIUM Phase II

Preliminary applications: Marviken CFT04

Input Parameters		Criteria	1	1	1	2	2	2	3	3	3	4	4	4	5	5	5
		P1	0	0.01	0.1	0	0.01	0.1	0	0.01	0.1	0	0.01	0.1	0	0.01	0.1
		P2	-	-	-	1	1.01	1.1	1	1.01	1.1	-	-	-	-	-	-
HF → W7	min		0.8	0.8	0.78	0.8	0.8	0.78	0.8	0.8	0.78	0.8	0.8	0.76	0.8	0.8	0.77
	max		0.86	0.86	0.87	0.86	0.86	0.87	0.86	0.86	0.87	0.87	0.87	0.87	0.87	0.87	0.87
HF → W8	min		0.14	0.14	<0.11	0.14	0.14	<0.11	0.14	0.14	<0.11	0.14	0.13	<0.11	0.14	0.13	<0.11
	max		0.34	0.36	0.43	0.34	0.36	0.43	0.34	0.36	0.43	0.43	0.43	0.43	0.43	0.43	0.43
Level → W5	min		0.4	<0.25	<0.25	0.4	<0.25	<0.25	0.4	<0.25	<0.25	0.37	<0.25	<0.25	0.4	<0.25	<0.25
	max		0.4	0.625	>0.95	0.4	0.625	>0.95	0.4	0.625	>0.95	0.95	>0.95	>0.95	0.42	>0.95	>0.95

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Objective of Specifications for PREMIUM Phase II

Preliminary applications: Marviken CFT04

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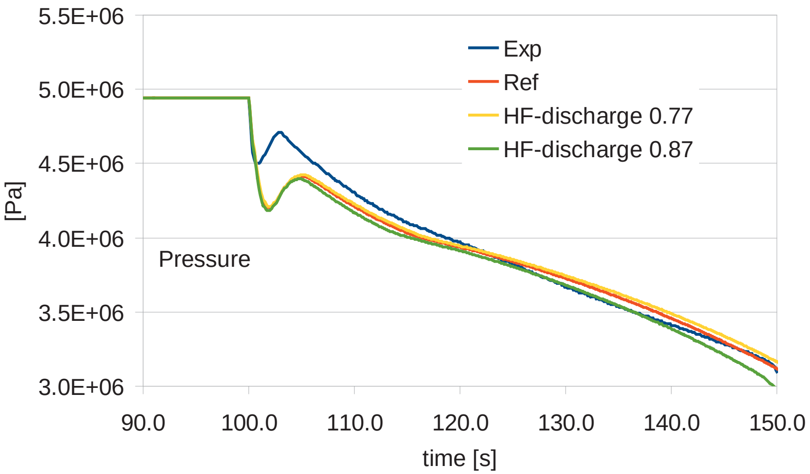
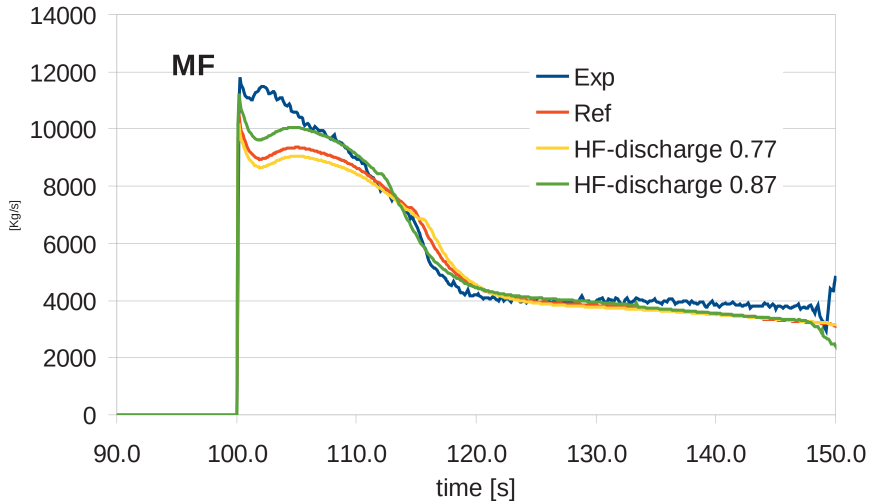
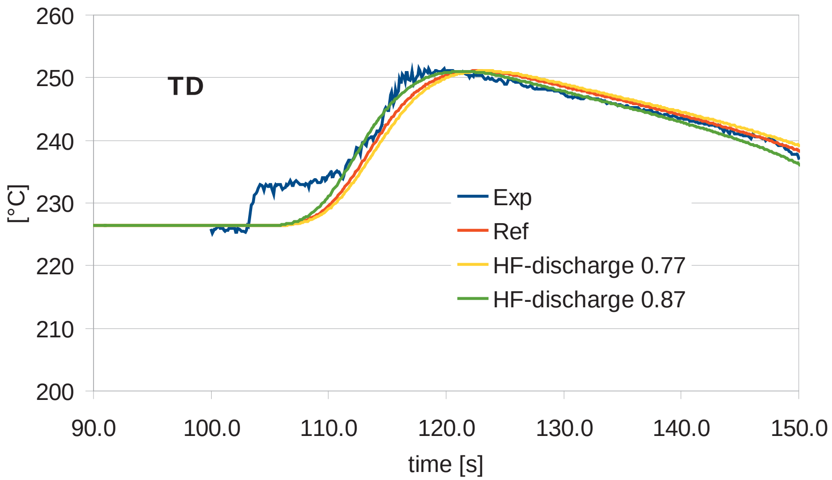
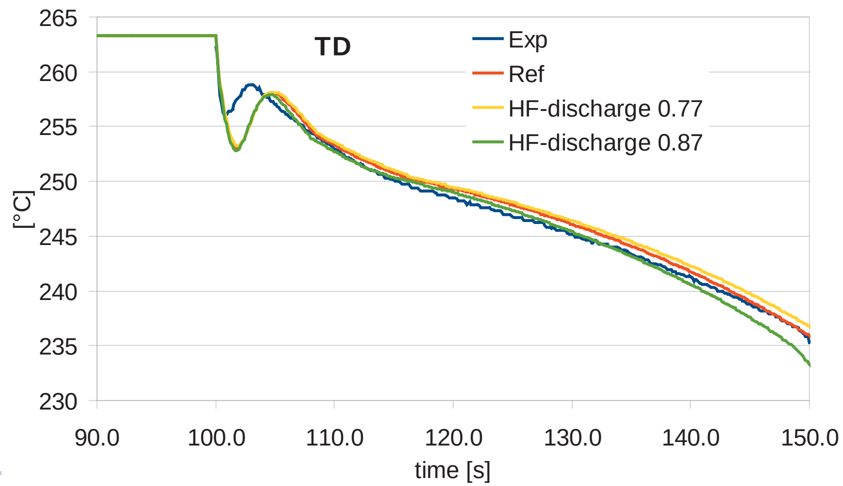
Calculation cases at the extremes of the Range of the Input Uncertainty Parameters







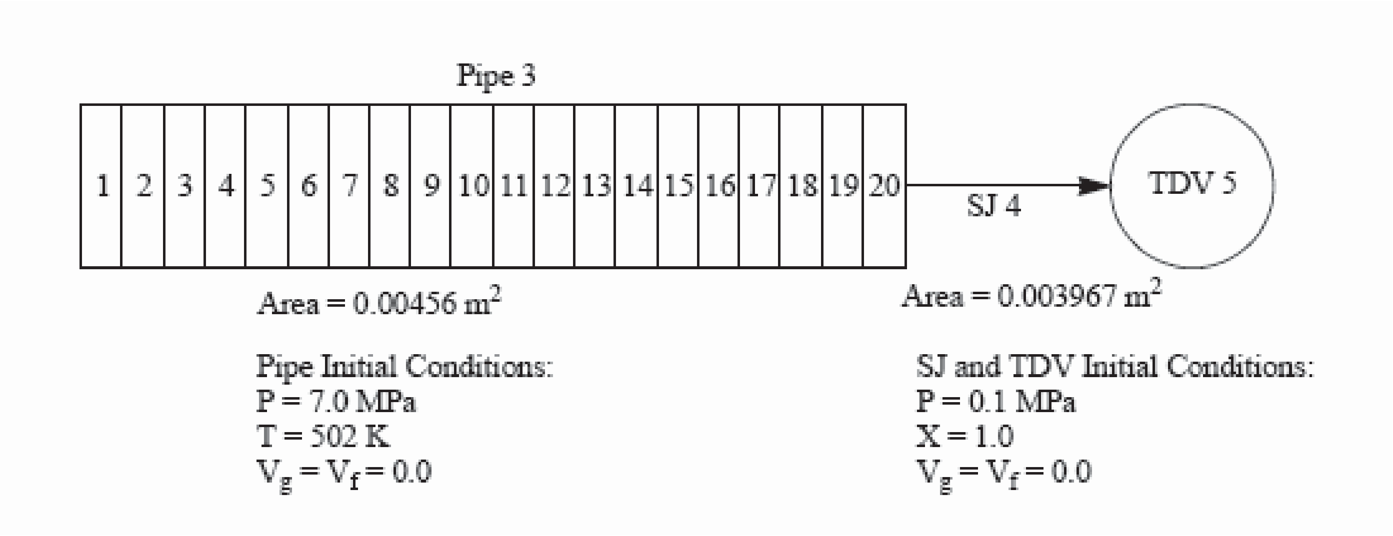


Objective of Specifications for PREMIUM Phase II

Preliminary applications: Edwards pipe

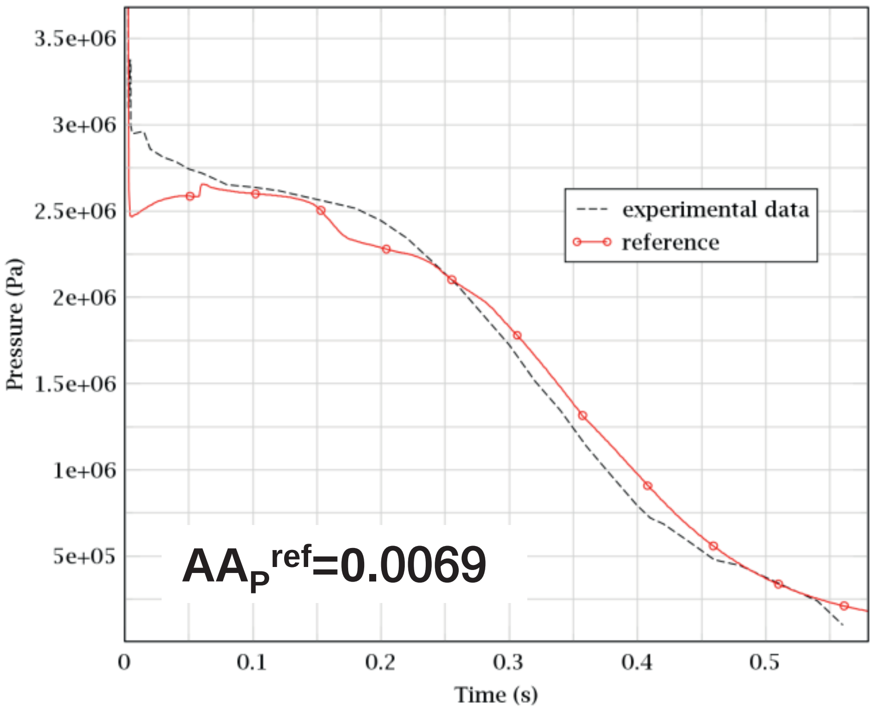
- **Selected responses:**
  - Pressure (P)
  - Void fraction (V)

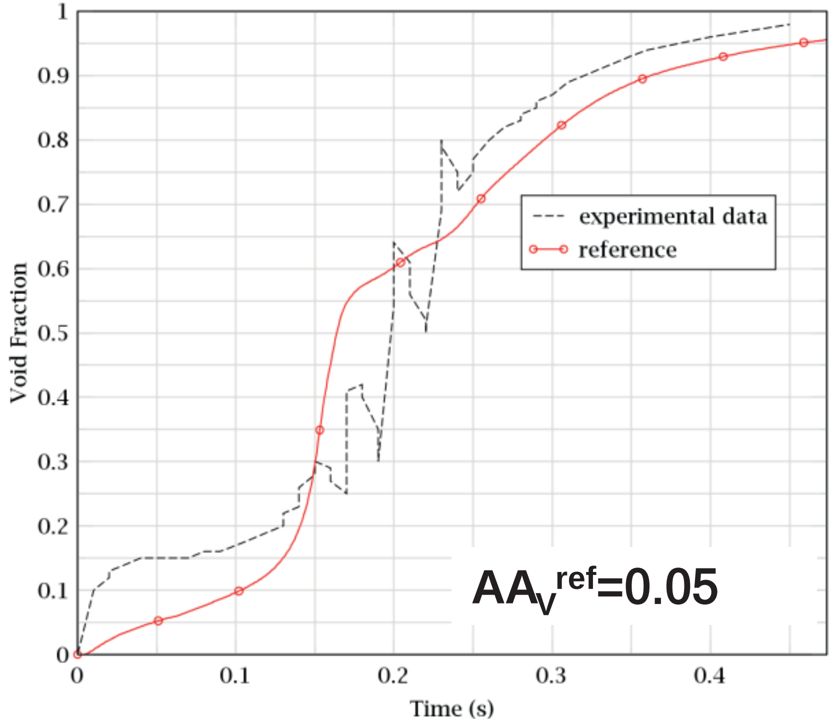
- **Set of Input Parameters (about 10)**



Objective of Specifications for PREMIUM Phase II

Preliminary applications: Edwards pipe





Objective of Specifications for PREMIUM Phase II

Preliminary applications: Edwards pipe

Selection of Input Parameter

1. Form loss coefficient (K_{loss})
2. Initial fluid temperature
3. Break area
4. “Henry-Fauske” choked flow model, discharge coefficient

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Objective of Specifications for PREMIUM Phase II

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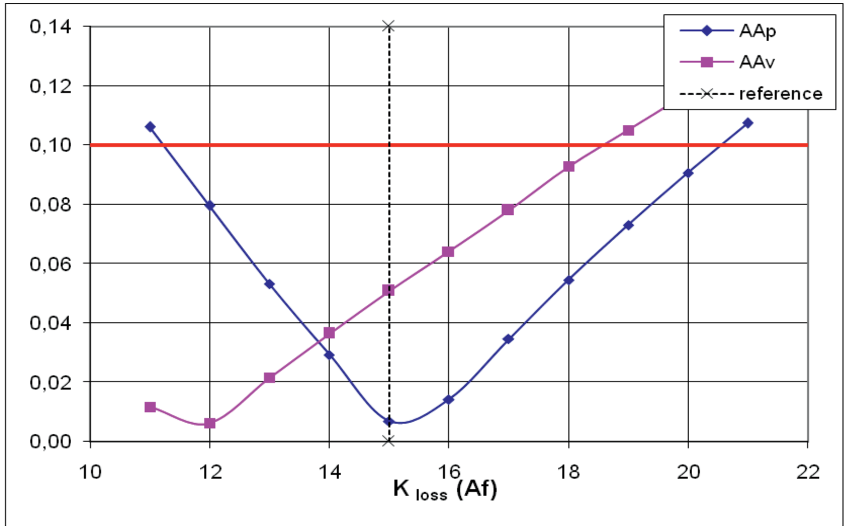
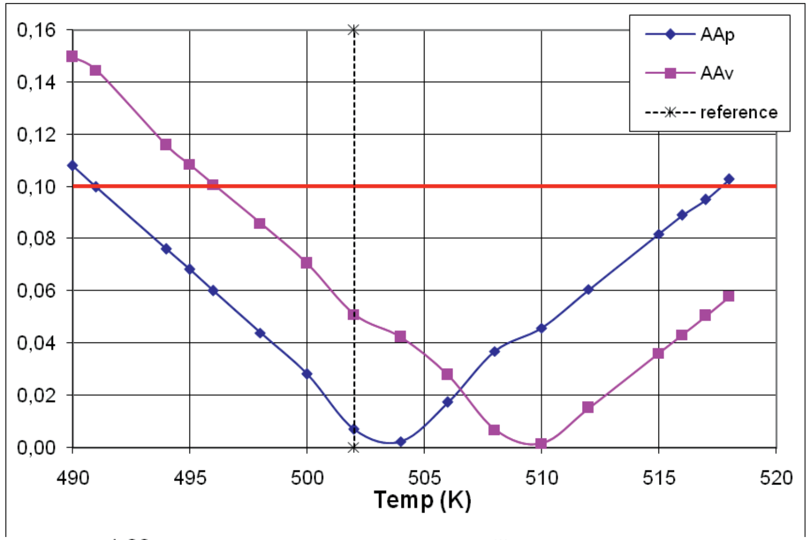
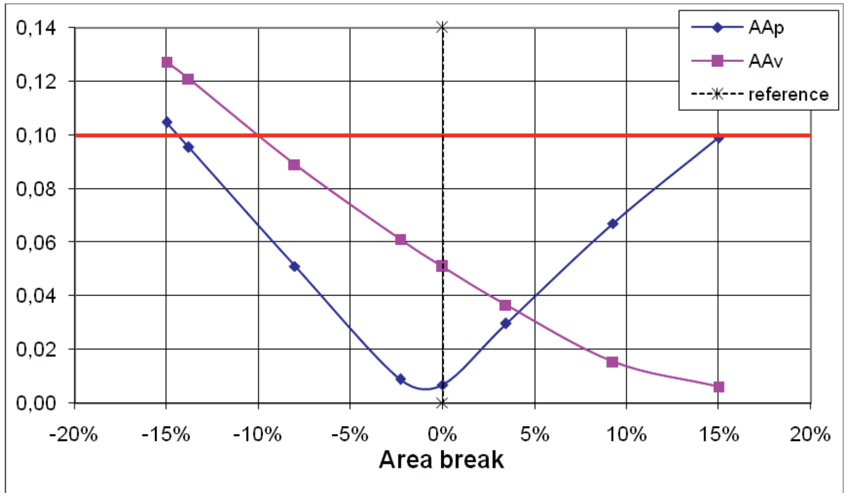
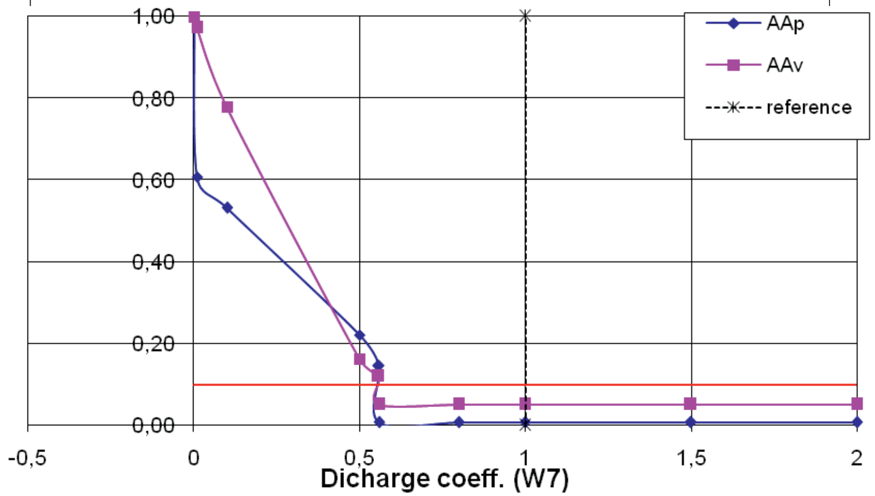
Preliminary applications: Edwards pipe

Results of FFTBM application









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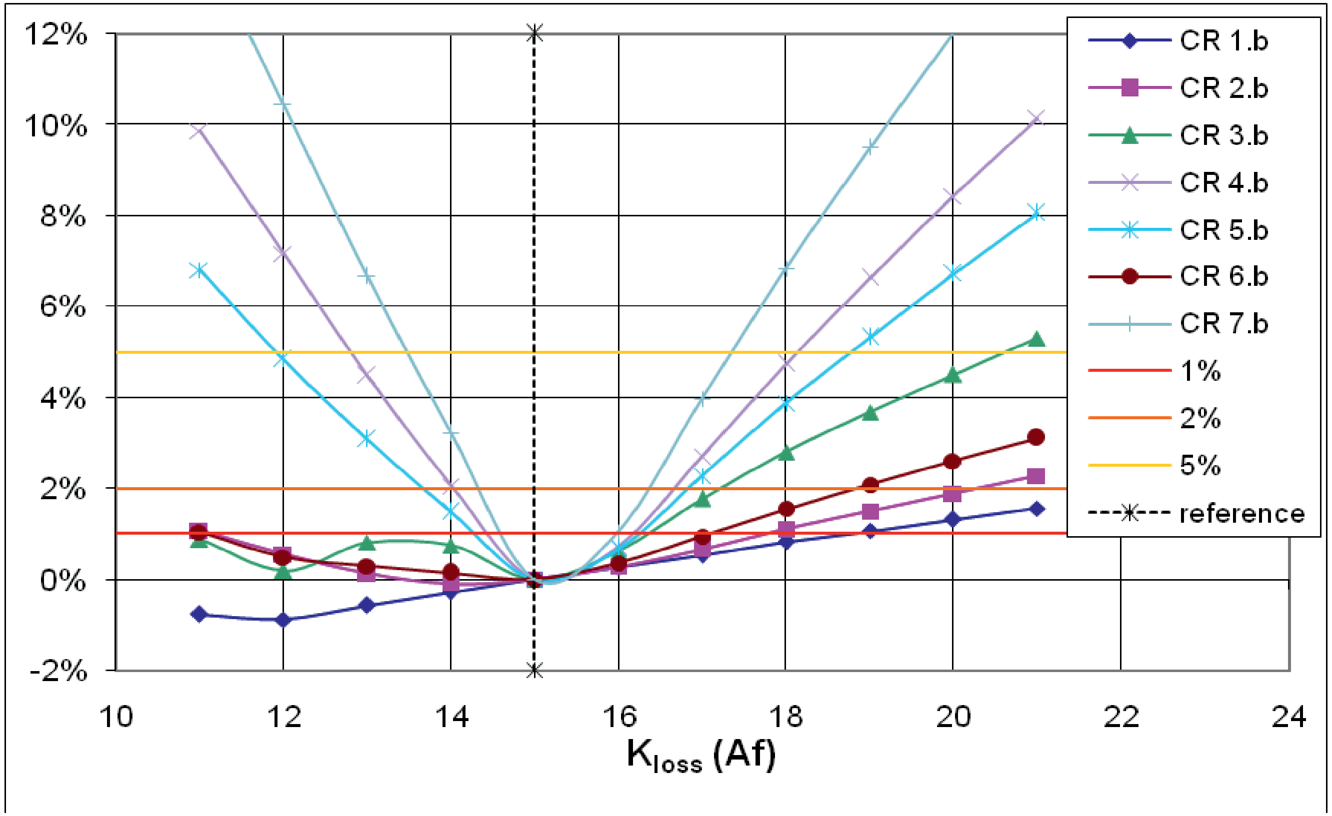
Objective of Specifications for PREMIUM Phase II

Preliminary applications: Edwards pipe

Application of criteria for K_{loss}



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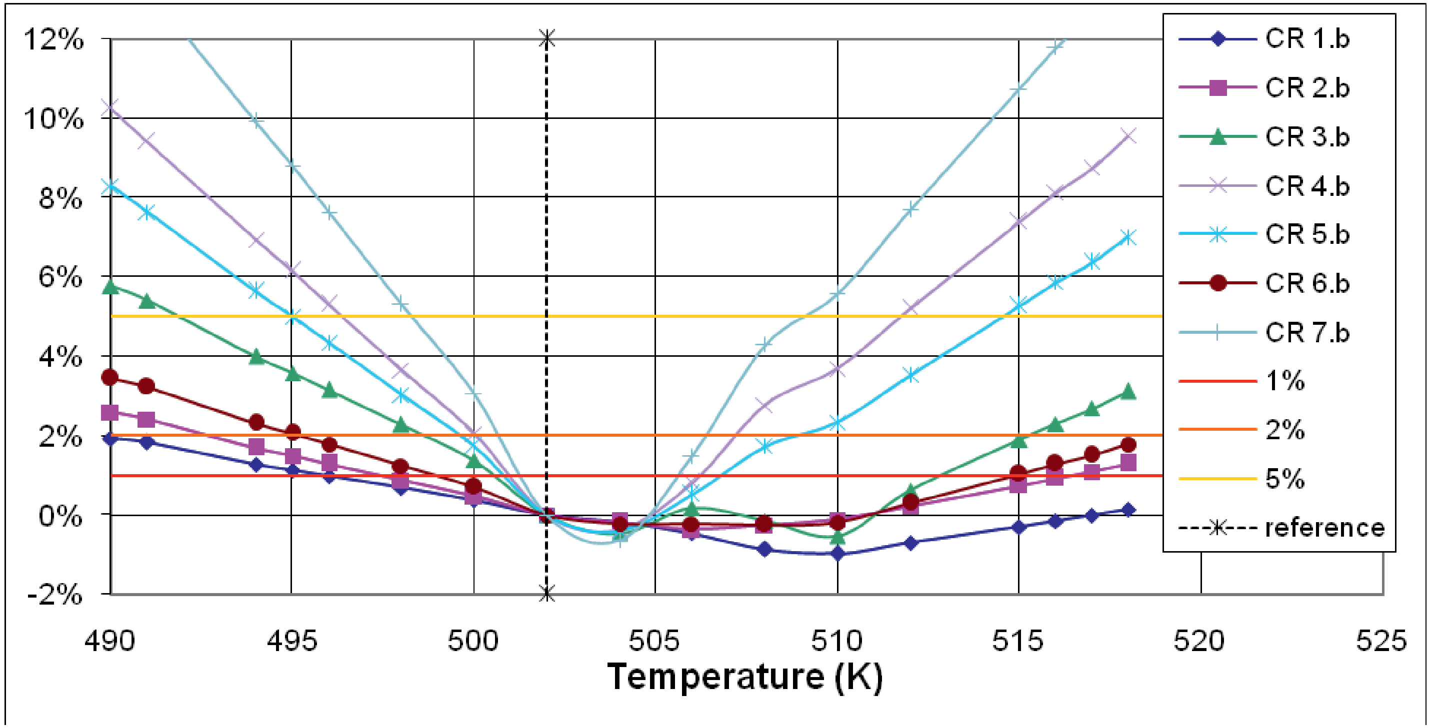
Objective of Specifications for PREMIUM Phase II

Preliminary applications: Edwards pipe

Application of criteria for initial fluid tempeature



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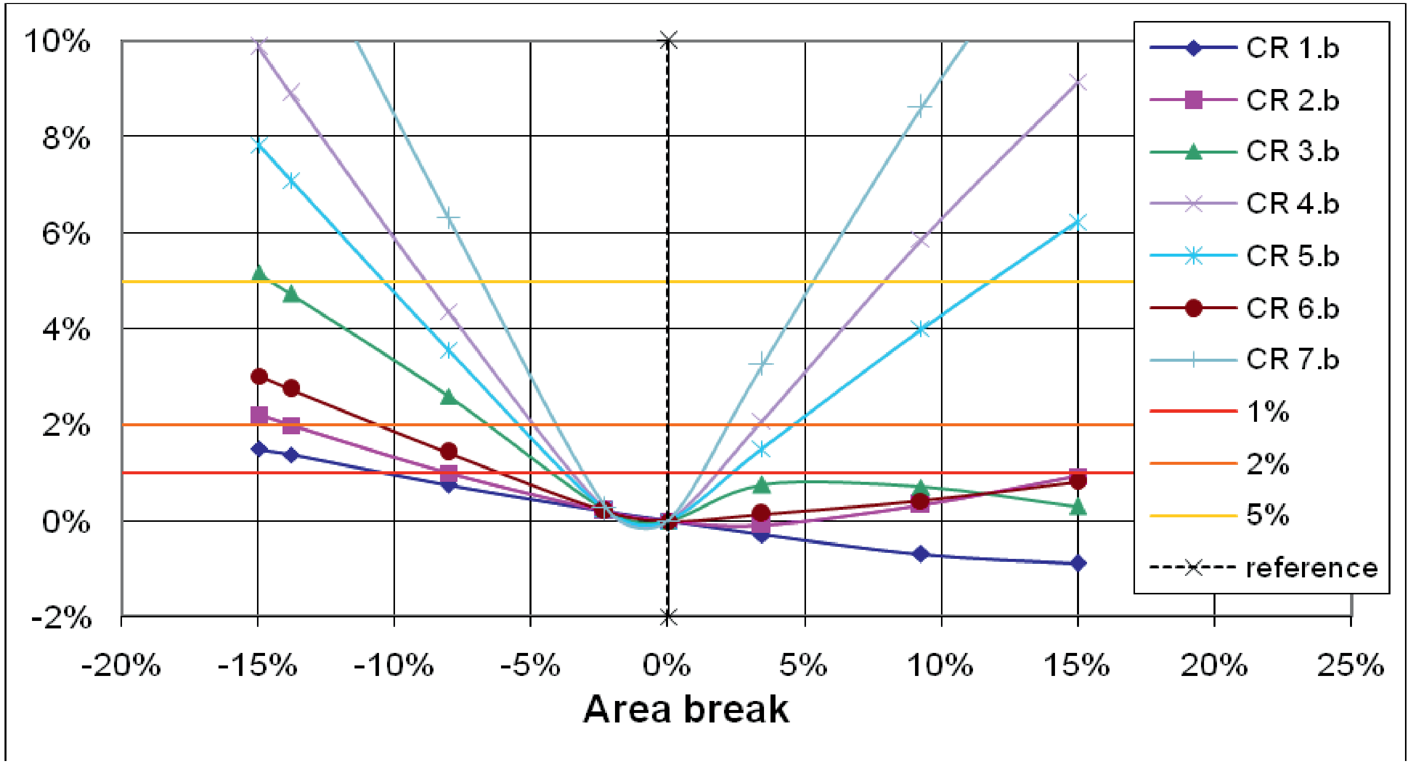
Objective of Specifications for PREMIUM Phase II

Preliminary applications: Edwards pipe

Application of criteria for break area



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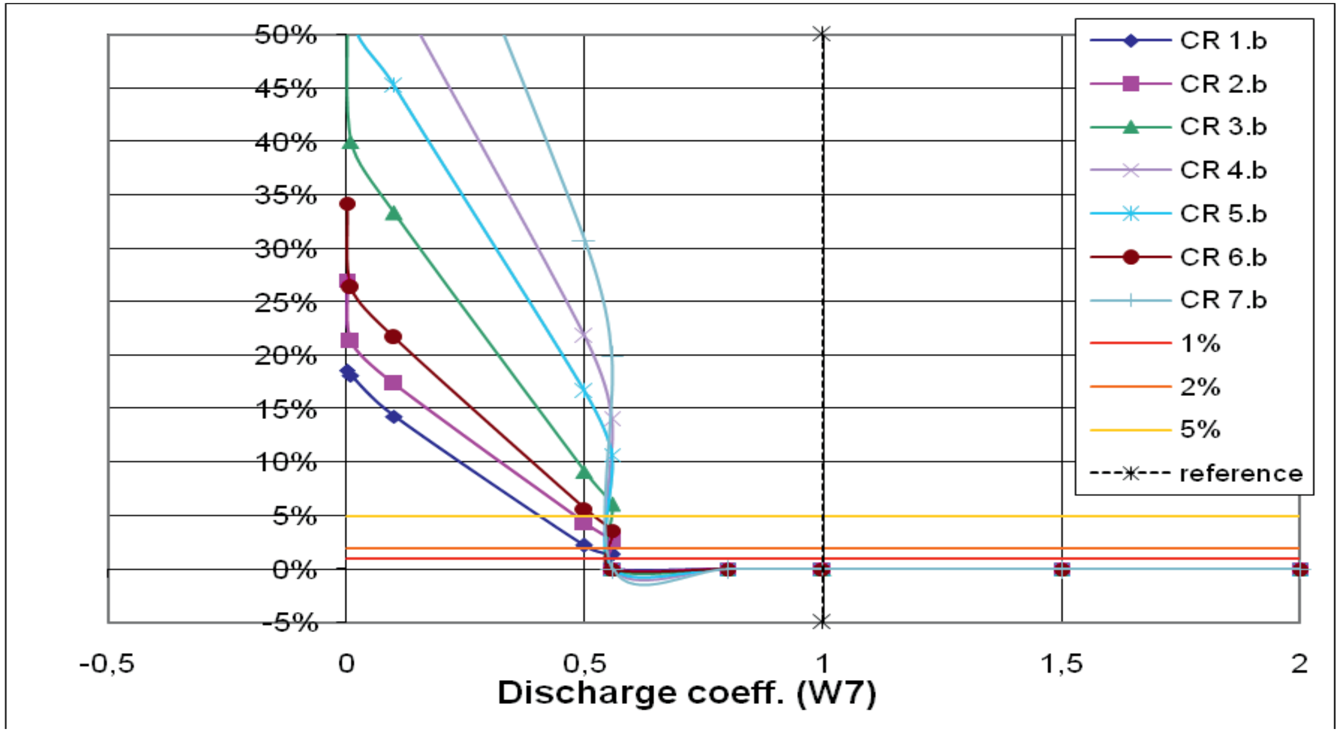
Objective of Specifications for PREMIUM Phase II

Preliminary applications: Edwards pipe

Application of criteria for discharge coefficient



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Objective of Specifications for PREMIUM Phase II

Preliminary applications: Edwards pipe

Input parameter variation margins

P 1		CR 1.b 1%	CR 1.b 2%	CR 1.b 5%	CR 2.b 1%	CR 2.b 2%	CR 2.b 5%	CR 3.b 1%	CR 3.b 2%	CR 3.b 5%
K_{loss}	min	11 (*)	Always (*)	Always (*)	11.15	11 (*)	Always (*)	11 (*)	11 (*)	11 (*)
	max	18.74			17.78	20.28		16.36	17.21	20.62
T(K)	min	495.9	Always (*)	Always (*)	497.4	492.8	Always (*)	500.6	498.7	491.9
	max	518 (*)			516.5	518 (*)		512.8	515.2	518 (*)
A_{break}(%)	min	- 10.4	Always (*)	Always (*)	- 8	- 13.8	Always (*)	- 4.2	- 6.6	- 14.5
	max	15 (*)			15 (*)	15 (*)		15 (*)	15 (*)	15 (*)
W7	min / max	> 0.559	> 0.508	> 0.404	> 0.559	> 0.559	> 0.479	> 0.559	> 0.559	> 0.559

P 1		CR 4.b 1%	CR 4.b 2%	CR 4.b 5%	CR 5.b 1%	CR 5.b 2%	CR 5.b 5%	CR 6.b 1%	CR 6.b 2%	CR 6.b 5%
K_{loss}	min	14.43	14.02	12.79	14.27	13.67	11.91	11.06	11 (*)	Always (*)
	max	16.18	16.67	18.13	16.25	16.83	18.78	17.09	18.86	
T(K)	min	501	500.1	496.4	500.8	499.6	495	498.9	495.2	Always (*)
	max	506.2	507.2	511.7	506.7	508.8	514.5	514.8	518 (*)	
A_{break}(%)	min	- 3.5	- 4.9	- 8.9	- 3.7	- 5.4	- 10.4	- 6.1	- 10.6	Always (*)
	max	1.8	3.3	7.9	2.3	4.5	11.8	15 (*)	15 (*)	
W7	min / max	> 0.559	> 0.559	> 0.559	> 0.559	> 0.559	> 0.559	> 0.559	> 0.559	> 0.559

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University of Pisa
DIMNP-GRNSPG
Nuclear Research Group in San Piero a Grado (Pisa)- Italy

The Findings from the OECD/NEA/CSNI UMS (Uncertainty Method Study)

F. D'Auria, H. Glaeser

OECD/CSNI Workshop on
"Best Estimate Methods and Uncertainty Evaluations"

Hosted by UPC & CSN
Barcelona, Spain, 16-18 November 2011

SCOPE & OBJECTIVE

1. **"The Uncertainty Methods Study (UMS – completed 1998) compares different methods to estimate the uncertainty in predictions of advanced best estimate thermal hydraulic codes by applying the methods to a particular experiment."**

2. **The results from the comparison are summarized considering recent evaluations and findings.**

3. **An outline of the milestones for the application of BEPU is given in advance.**

HISTORIC OUTLINE, 50's & 60's

50's

Accidents and related scenarios in nuclear power plants were considered to demonstrate the safety of NPP when computers did not exist. Experiments, pioneering thermal-hydraulics models and engineering evaluations were the basis of the reactor safety analyses.

60's

More systematic thermal-hydraulic studies and experiments were conducted, noticeably concerning individual 'physical' phenomena like TPCF, CHF, Depressurization/Blow-down, etc.

HISTORIC OUTLINE, 70's

Massive use of computers for nuclear reactor safety started. The AA could benefit of primitive numerical codes and of results of lately called integral-system experiments.

- 'Interim Acceptance Criteria for ECCS' in 1971.
- The Appendix K to the paragraph 10-CFR-50.46 in 1974.
- 'Conservatism' is the key-word.
- WASH-1400 or the "Rasmussen Report" was issued addressing the relevance of PSA.

HISTORIC OUTLINE, 80's

Robust, user-friendly versions of lately called system-thermal-hydraulic codes were available.

- **The importance of V & V became clear.**
- **'The scaling issue' came.**
- **CSNI proposed viable ways for V & V involving the evaluation of the UE, and the recognition of the role of the Nodalization (N) and related qualification (CCVM & SOAR on TECC).**
- **App. K continued to be used for licensing purposes.**

HISTORIC OUTLINE, 90's

The need for uncertainty (U) evaluation became clear.

- **Working approaches for U were proposed, e.g.**
  - **CSAU by USNRC, 1989,**
  - **GRS <Wilks formula>, 1990,**
  - **UMAE by Un. Pisa, <accuracy extrapolation>, 1993 (bases in 1988).**
- **UMS project was carried out**
- **USNRC issued RG 1.157: BE codes allowed with conservatism in models and BIC.**
- **The acronym BEPU was proposed.**
- **Tools available 'to quantify' the qualification level of Code and of Nodalization.**
- **App. K continued to be used for licensing purposes.**

HISTORIC OUTLINE, 00's – 1 of 2

Applications of BEPU approaches in licensing processes definitely started. Key events:

a) **CIAU** (Code with capability of Internal Assessment of Uncertainty) **method issued in 2000, following the break-through Meeting of Annapolis in 1996.**

b) **BEPU LBLOCA analysis for Angra-2 NPP licensing, 2002, by Framatome-AREVA.**

c) **USNRC issued the RG 1.203.**

d) **CSNI launched and completed the six-year project BEMUSE.**

e) **IAEA issued SRS reports No. 23 & 52.**

HISTORIC OUTLINE, 00's – 2 of 2

Applications of BEPU approaches in licensing processes definitely started. Key events, cont. ed:

f) ANS Conferences BE-2000 and BE-2004 were held. V & V Workshops were held in Idaho Falls (Id, US) and in Myrtle Beach (NC, US).

g) A variety of BEPU industrial applications, e.g. ASTRUM by Westinghouse (license renewal and power up-rating framework) were submitted.

h) Bifurcation analysis possible (by using CIAU).

i) BEPU Chapter 15 analyses for Atucha-2 NPP licensing, 2002, by NA-SA & Univ. Pisa (2010).

OBJECTIVE OF UMS

1. To gain insights into differences between features of the methods by:

   - comparing the different methods, step by step, when applied to the same problem;
   - comparing the uncertainties predicted for specified output quantities of interest;
   - comparing the uncertainties predicted with measured value;
   - and so allowing conclusions to be drawn about the suitability of methods.

2. To inform those who will take decisions on conducting uncertainty analyses, for example in the light of licensing requirements.

METHODS ADOPTED IN UMS

Participant	Code Version Used	Method Name and Type
AEA Technology, UK	RELAP5/MOD3.2	AEAT Method. Phenomena uncertainties selected, quantified by ranges and combined.
University of Pisa, Italy	RELAP5/MOD2 cycle 36.04, IBM version CATHARE 2 version 1.3U rev 5	Uncertainty Method based on Accuracy Extrapolation (UMAE). Accuracy in calculating similar integral tests is extrapolated to plant.
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany	ATHLET Mod 1.1 Cycle A	GRS Method. Phenomena uncertainties quantified by ranges and probability distributions (PDs) and combined.
Institut de Protection et de Sûreté Nucléaire (IPSN), France	CATHARE 2 version 1.3U rev 5	IPSN Method. Phenomena uncertainties quantified by ranges and PDs and combined.
Empresa Nacional del Uranio, SA (ENUSA), Spain	RELAP5/MOD 3.2	ENUSA Method. Phenomena uncertainties quantified by ranges and PDs and combined.

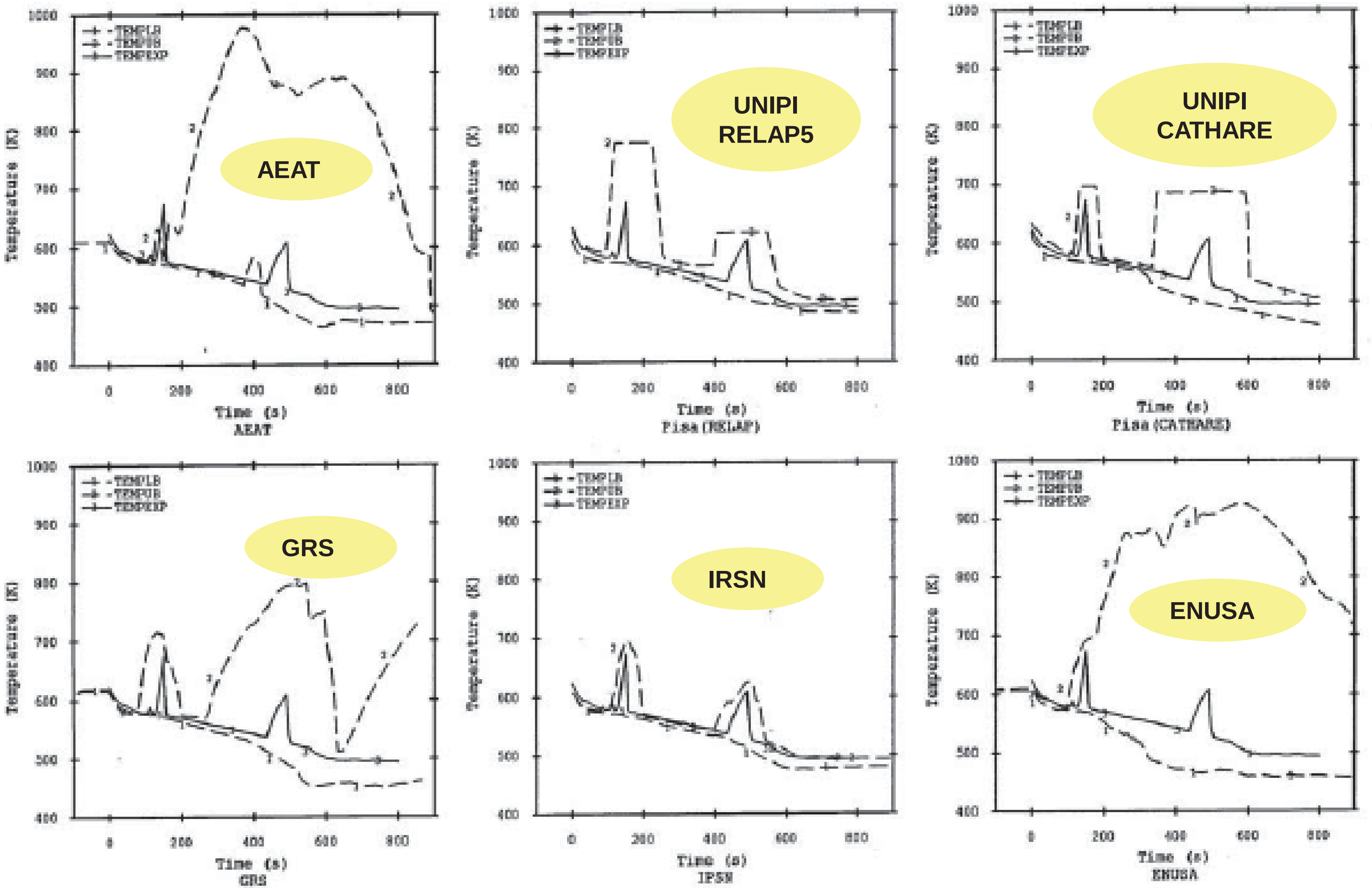
THE UMS EXPERIMENT

SBLOCA-BDBA INCLUDING THREE 'POTENTIAL' DNB SITUATIONS

1. **LOOP SEAL DRIVEN AT H-PRESSURE** (loop seal clearing 'quenches' the RST excursion).
2. **MASS DEPLETION AT M-PRESSURE** (accumulator intervention 'quenches' the RST excursion).
3. **MASS DEPLETION AT L-PRESSURE** (eventually, quenched by <late> actuation of LPIS).

- **Time duration of about 600 s <10'>**
- **Challenging for U methods:** 'setting' parameters to predict U associated with one DNB may affects U prediction for subsequent DNB

THE UMS KEY RESULTS – 1 OF 2



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THE UMS KEY RESULTS – 2 OF 2

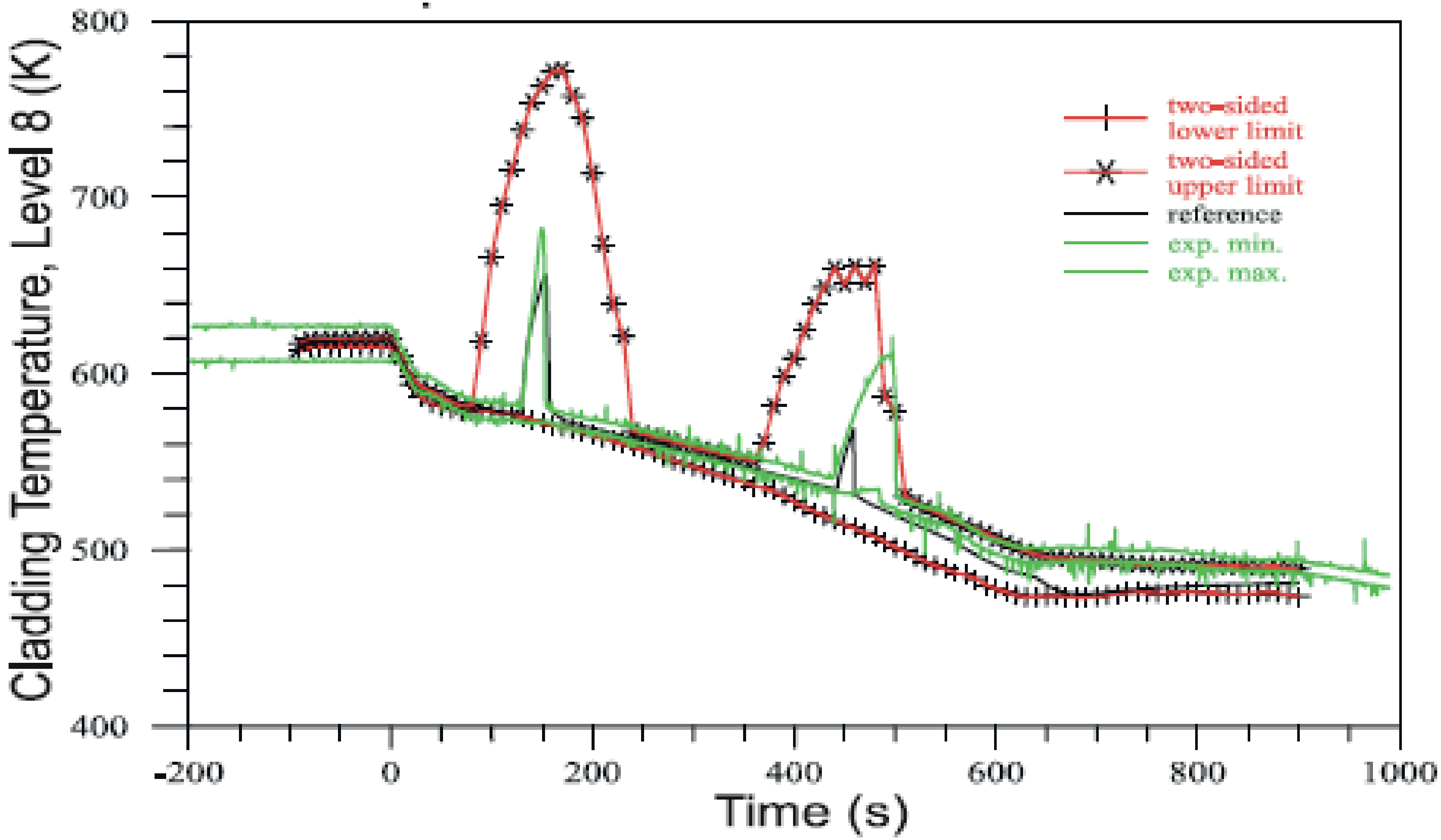
ALL PARTICIPANTS CALCULATED THAT ‘UNCERTAINTY BAND THICKNESS’ INCREASES AND DECREASES DURING THE CALCULATED TRANSIENT

THIS RAISES THE ISSUE OF DISTINGUISHING BETWEEN ‘TIME UNCERTAINTY’ AND ‘QUANTITY UNCERTAINTY’:

- **‘time uncertainty’ (or error in predicting the time of occurrence of any event) should not decrease with time.**
- **‘quantity uncertainty’ (e.g. error in predicting mass inventory) may be larger during the fast depressurization and ‘small’ at the end of blow-down.**

POST-UMS RESULTS – 1 of 4 –

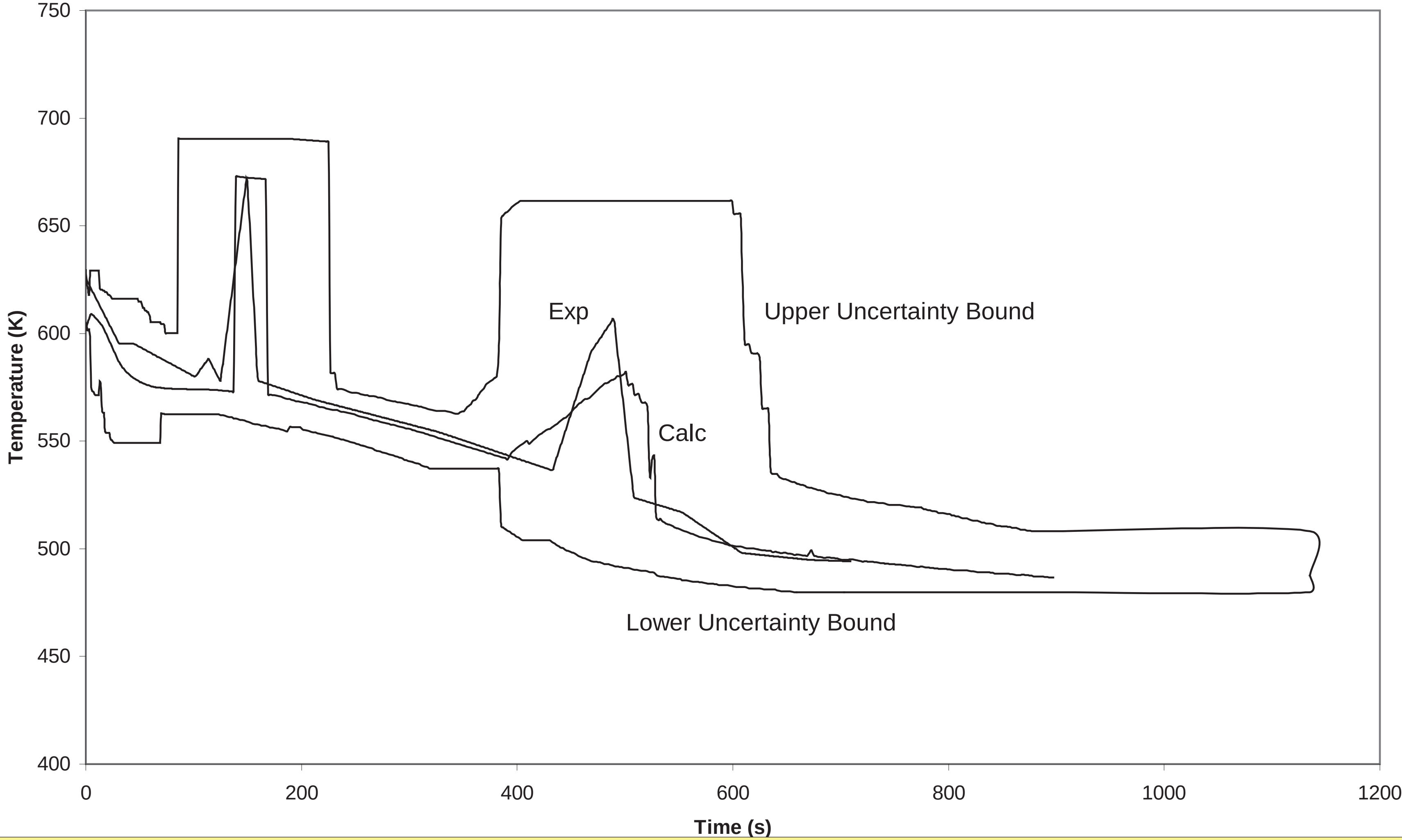
GRS SUBMITTTED UP-DATED RESULTS



Uncertainty bands 'more-similar' to results from UMAE application

POST-UMS RESULTS – 2 of 4 –

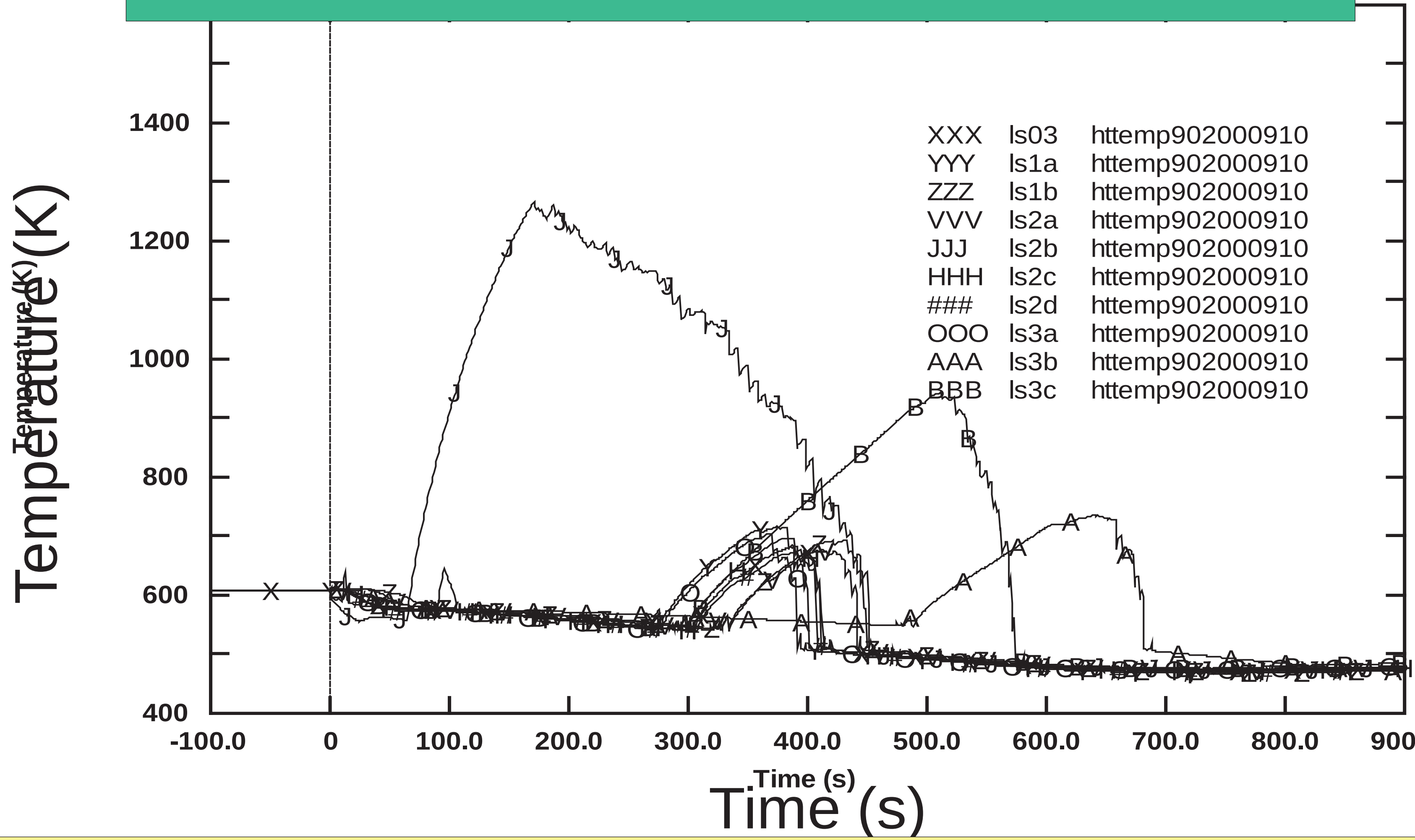
CIAU WAS APPLIED TO UMS



Calculated 'automatic' uncertainty bands confirm the UMAE results

POST-UMS RESULTS – 3 of 4 –

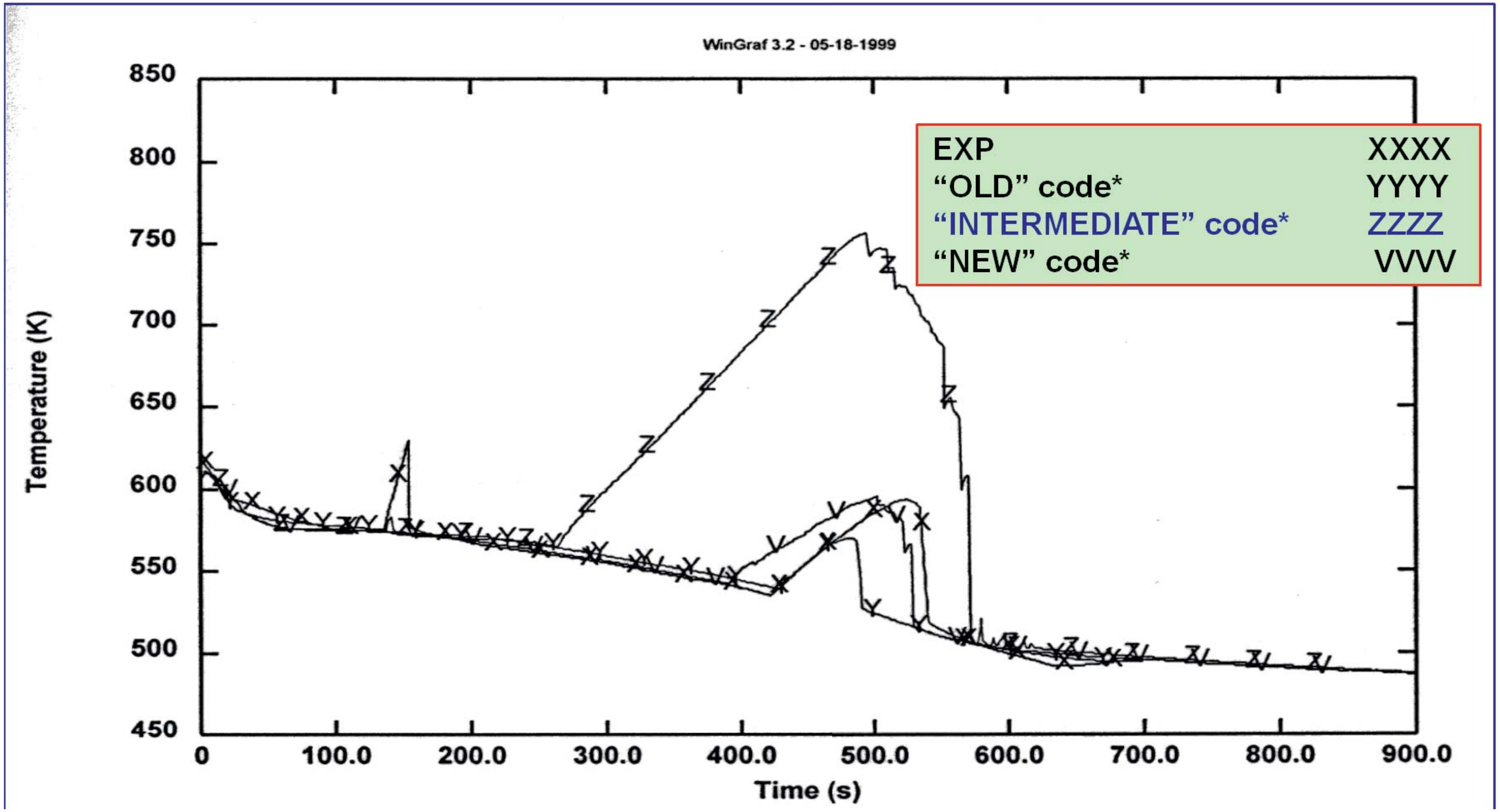
BIFURCATION ANALYSIS PERFORMED IN RELATION TO THE UMS PROBLEM



Results confirm the AEAT and the ENUSA results with 'low' probability of occurrence (see also next)

POST-UMS RESULTS – 4 of 4 –

CONSIDERATION OF CODE VERSION



The 'intermediate' code was used by AEAT and by ENUSA. This might contribute explaining the results by those two participants

CONCLUSIONS – 1 of 2

1. Significant differences in calculating the wideness of the time-dependent uncertainty bands: this may cause misleading conclusion.

2. Large band wideness calculated by AEAT and ENUSA may raise concerns related to the capability of codes and their applicability to the prediction of NPP transients.

3. In contrast, very small band wideness is calculated by IPSN: this is justified by the ‘reduced’ number of input uncertain parameters.

CONCLUSIONS - 2 of 2

4. The set of results calculated by UMAE (2 UMS calculations), by GRS (post-UMS calculation) and by CIAU (post-UMS calculation), show similar results and are consistent with the current capabilities of codes. These might be considered as reference results from the UMS.

5. Follow-up of UMS are BEMUSE (completed 2010) and PREMIUM (started 2011) projects.

6. It could be of interest ‘to repeat’ UMS with input uncertain parameters and ranges of variations selected following a “*deterministic*” procedure.

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OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations

MAIN RESULTS OF THE OECD BEMUSE PROGRAMME

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ACKNOWLEDGMENTS

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UPC's team: Marina Pérez, Lluís Batet and Raimon Pericas

Summary

1. Objectives of the programme
2. Main steps
3. Used methods
4. Selected results
   - 4.1 Application to LOFT L2-5 experiment
   - 4.2 Application to Zion nuclear power plant
5. Conclusions and recommendations

Summary

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1. Objectives of the programme

Background

The **conservative codes** contain assumptions to try to cover not known uncertainties. These assumptions are often unphysical and lead to predictions that could be far from reality

BE codes are designed to model all the relevant processes in a physically realistic. A calculation with a BE code is then considered the best approach of what is more likely to occur.

In any case it is necessary to evaluate the **uncertainty** of the estimation

1. Objectives of the programme

Background

In the near past under the auspices of CSNI, the comparative exercise called **UMS** (Uncertainty Methods Study) has been launched on uncertainty methodologies used for thermal-hydraulic best-estimate codes

More recently (from 2003) the OECD **BEMUSE** started with the aim of achieving a deeper understanding such methods

1. Objectives of the programme

Objectives

The objectives of this programme are:

- To evaluate the **practicability, quality and reliability** of best-estimate methods including uncertainty evaluations in applications relevant to nuclear reactor safety.
- To develop **common** understanding.
- To promote / facilitate their use by the **regulator bodies** and the **industry**

1. Objectives of the programme

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Objectives

The BEMUSE programme is focussed on the application of uncertainty methodologies to **Large Break LOCAs**

Using the similar codes and similar methods should allow comparing the **potential important uncertain parameters** and the **effects of different** modelling for uncertainties can be evaluated

Therefore, the **assessment of each methodology** by comparison with experimental data is also one of the purposes of the programme.

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Summary

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2. Main steps

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The BEMUSE program is divided in two steps:

1 - Uncertainty and sensitivity analysis of **LOFT L2-5** test calculations

2 - To perform this analysis for a **NPP-LB**

2. Main steps

First step (Phases 1, 2 and 3):

Phase I: presentation a priori of the uncertainty evaluation methodology to be used (lead organisation: IRSN)

Phase II: re-analysis of the ISP-13 exercise, post-test of LOFT L2-5 test (lead organisation: University of Pisa)

Phase III: uncertainty evaluation of the L2-5 test calculations (lead organisation: CEA)

2. Main steps

Second step (Phases 4, 5 and 6):

Phase IV: best-estimate analysis of an NPP-LBLOCA (lead organisation: UPC)

Phase V: uncertainty evaluation of the NPP-LBLOCA (lead organisation: UPC)

Phase VI: status report, conclusions and recommendations (lead organisation: GRS)

Summary

3. Used methods

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Participants and used codes

No.	Organisation	Country	Code	Participation in Phases
1	AEKI	Hungary	ATHLET2.0A	1, 2, 4, 5
2	CEA	France	CATHARE2V2.5_1	1, 2, 3, 4, 5
3	EDO “Gidropress“	Russia	TECH-M-97	2, 4, 5
4	GRS	Germany	ATHLET1.2C/ 2.1B	1, 2, 3, 4, 5
5	IRSN	France	CATHARE2V2.5_1	1, 2, 3, 4, 5
6	JNES	Japan	TRACE ver4.05	1, 2, 3, 4, 5
7	KAERI	South Korea	MARS 2.3/ 3.1	2, 3, 4, 5
8	KINS	South Korea	RELAP5 mod3.3	1, 2, 3, 4, 5
9	NRI-1	Czech Republic	RELAP5 mod3.3	2, 3, 4, 5
10	NRI-2	Czech Republic	ATHLET2.0A/ 2.1A	1, 2, 3, 5
11	PSI	Switzerland	TRACE v4.05 5rc3	1, 2, 3, 4, 5
12	UNIPI-1	Italy	RELAP5 mod3.2	1, 2, 3, 4, 5
13	UNIPI-2	Italy	CATHARE2V2.5_1	4, 5
14	UPC	Spain	RELAP5 mod3.3	1, 2, 3, 4, 5

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3. Used methods

9 out of 10 participants adopt an uncertainty methodology based on a **propagation of input uncertainties**

These 9 organisations have chosen to follow a **probabilistic methodology**. All these methods have a lot of common characteristics (the use of order statistics / Wilks' formula)

The Pisa University is using its own method **CIAU-UMAE**, based on extrapolation of accuracy

It must be noted that **no participants** have used a deterministic method.

3. Used methods

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Probabilistic methods follow the three main steps:

a) Determination of the **Probability Density Functions**

b) **Propagation** of uncertainties

c) Determination of **response** uncertainty ranges

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3. Used methods

The CIAU method is based on the principle that it is reasonable to extrapolate code output errors observed for relevant experimental tests to real plants

The development of the method implies the **availability of qualified experimental data**

Steps: to check the quality of code results with respect to experimental data / to determine both Quantity Accuracy Matrix and Time Accuracy Matrix / to estimate 'time-domain' and 'phase-space' uncertainties for the considered scenario

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4. Selected results
4.1 Application to LOFT L2-5 experiment
Thermalhydraulic aspects

Phase II is the re-analysis of the ISP-13 exercise, post-test calculation of LOFT L2-5 test

The coordinator is the University of Pisa

LOFT L2-5 is a Large Break LOCA

LOFT 50-MWt PWR with instrumentation to measure and provide data on the TH and nuclear conditions

Operation of the LOFT system is typical of large (~1000 MWe) commercial PWR operations

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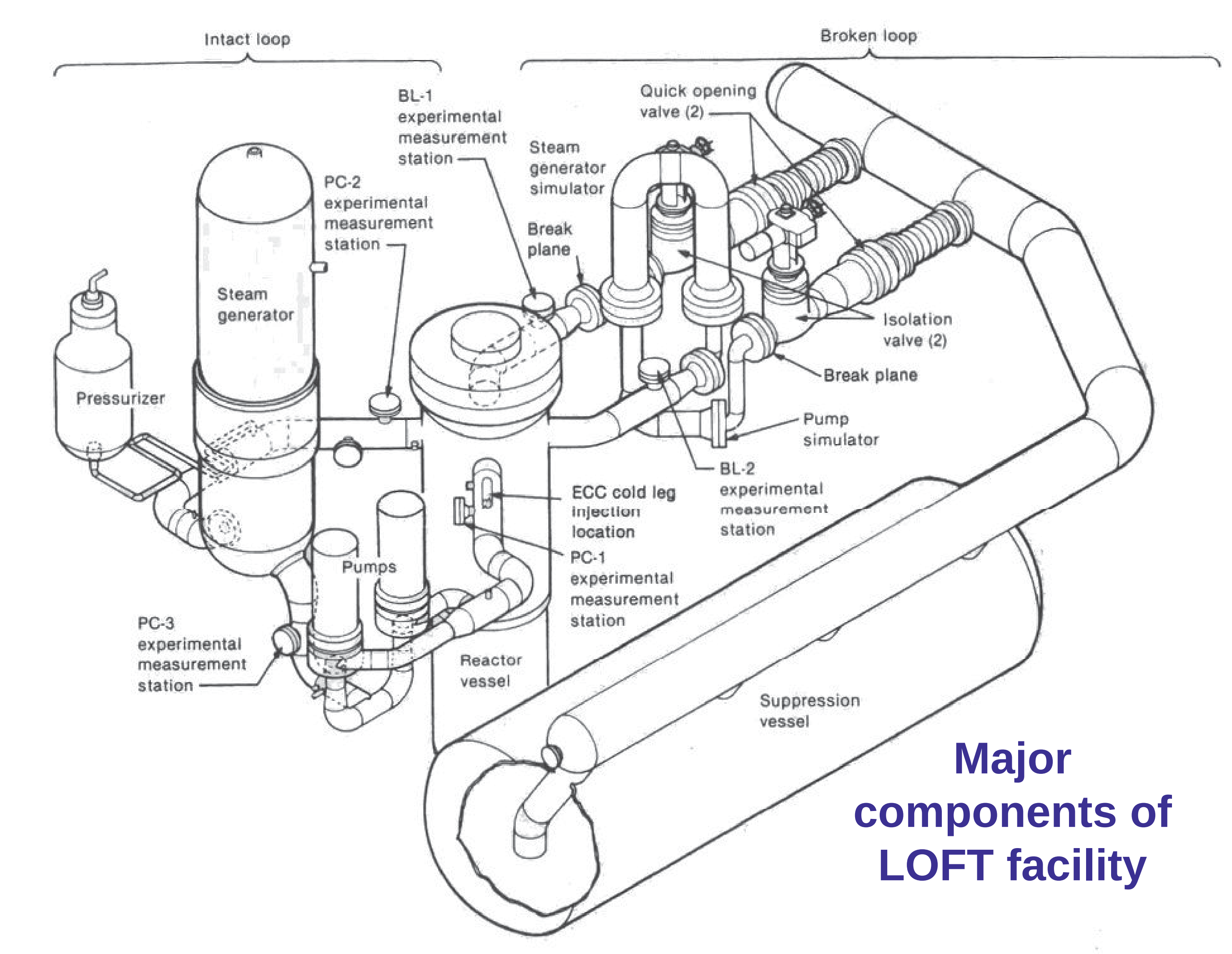
4. Selected results
4.1 Application to LOFT L2-5 experiment
Thermalhydraulic aspects



Major components of LOFT facility

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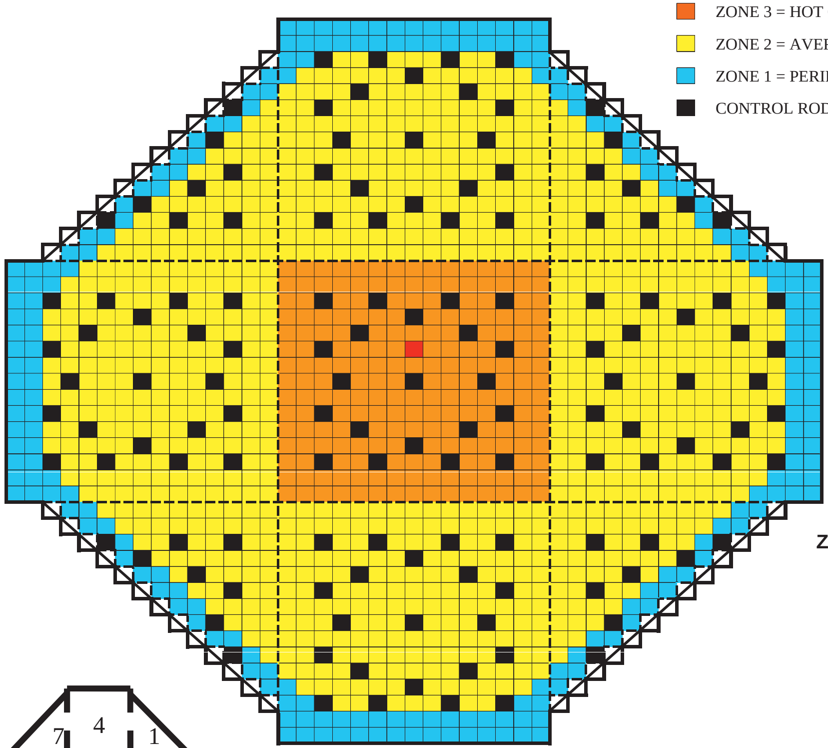
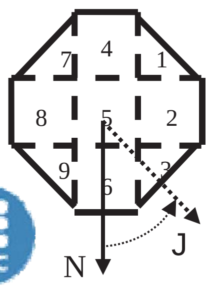
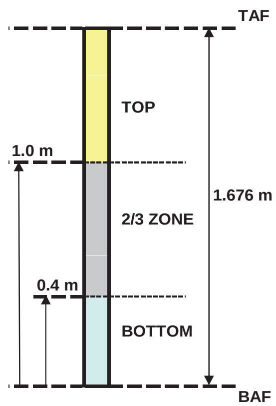
4. Selected results
4.1 Application to LOFT L2-5 experiment
Thermalhydraulic aspects

Core Geometry

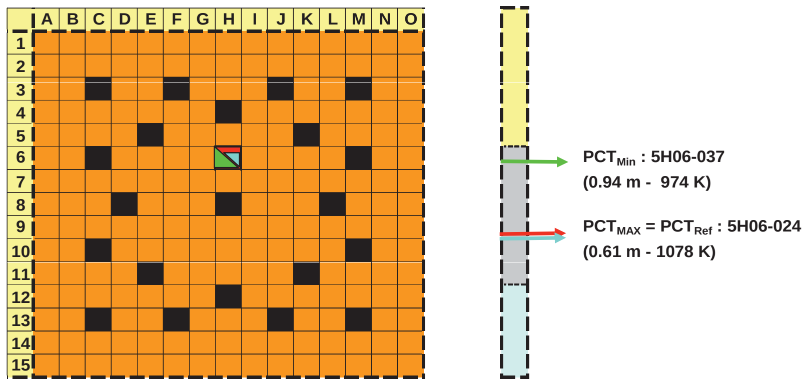
ZONE 4 = HOT ROD (RODS N° = 1)
ZONE 3 = HOT CHANNEL (RODS N° = 203)
ZONE 2 = AVERAGE CHANNEL (RODS N° = 876)
ZONE 1 = PERIPHERAL CHANNEL (RODS N° = 220)
CONTROL RODS (RODS N° = 137)







ZONE 4: HOT ROD (FUEL ASSEMBLY N° 5) - HEIGHT: 2/3



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4. Selected results

4.1 Application to LOFT L2-5 experiment

Thermalhydraulic aspects

Sequence of events of the test

EVENTS	TIME (s)
Experiment L2-5 initiated – break opening	0.0
Subcooled blowdown ended	0.043 $\pm$ 0.01
Reactor scrammed	0.24 $\pm$ 0.01
Cladding temperatures initially deviated from saturation	0.91 $\pm$ 0.2
Primary coolant pumps tripped	0.94 $\pm$ 0.01
Subcooled break flow ended (cold leg)	3.4 $\pm$ 0.5
Partial rewet initiated	12.1 $\pm$ 1.0
Pressurizer emptied	15.4 $\pm$ 1
Accumulator A injection initiated	16.8 $\pm$ 0.1
Partial rewet ended	22.7 $\pm$ 1.0
HPIS injection initiated	23.90 $\pm$0.02
Maximum cladding temperature reached (1078 K)	28.47 $\pm$ 0.02
LPIS injection initiated	37.32 $\pm$ 0.02
Accumulator emptied	49.6 $\pm$ 0.1
Core cladding quenched	65 $\pm$ 2
BST maximum pressure reached	72.5 $\pm$ 1
LPIS injection terminated (s)	107.1 $\pm$ 0.4

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4. Selected results
4.1 Application to LOFT L2-5 experiment
Thermalhydraulic aspects

A CONSISTENT CODE QUALIFICATION PROCESS BASED ON UMAE CRITERIA HAS BEEN APPLIED TO PHASE 2 OF BEMUSE

NODALIZATION QUALIFICATION

- Nodalization Tables
- Pressures Vs Length Curve

QUALITATIVE ACCURACY EVALUATION

- Resulting Time Sequence of Events
- Relevant Thermalhydraulic Aspects (RTA)
- Experimental Time Trends Comparisons – Qualitative Judgments

QUANTITATIVE ACCURACY EVALUATION

- Application of the FFTBM

4. Selected results
4.1 Application to LOFT L2-5 experiment
Thermalhydraulic aspects

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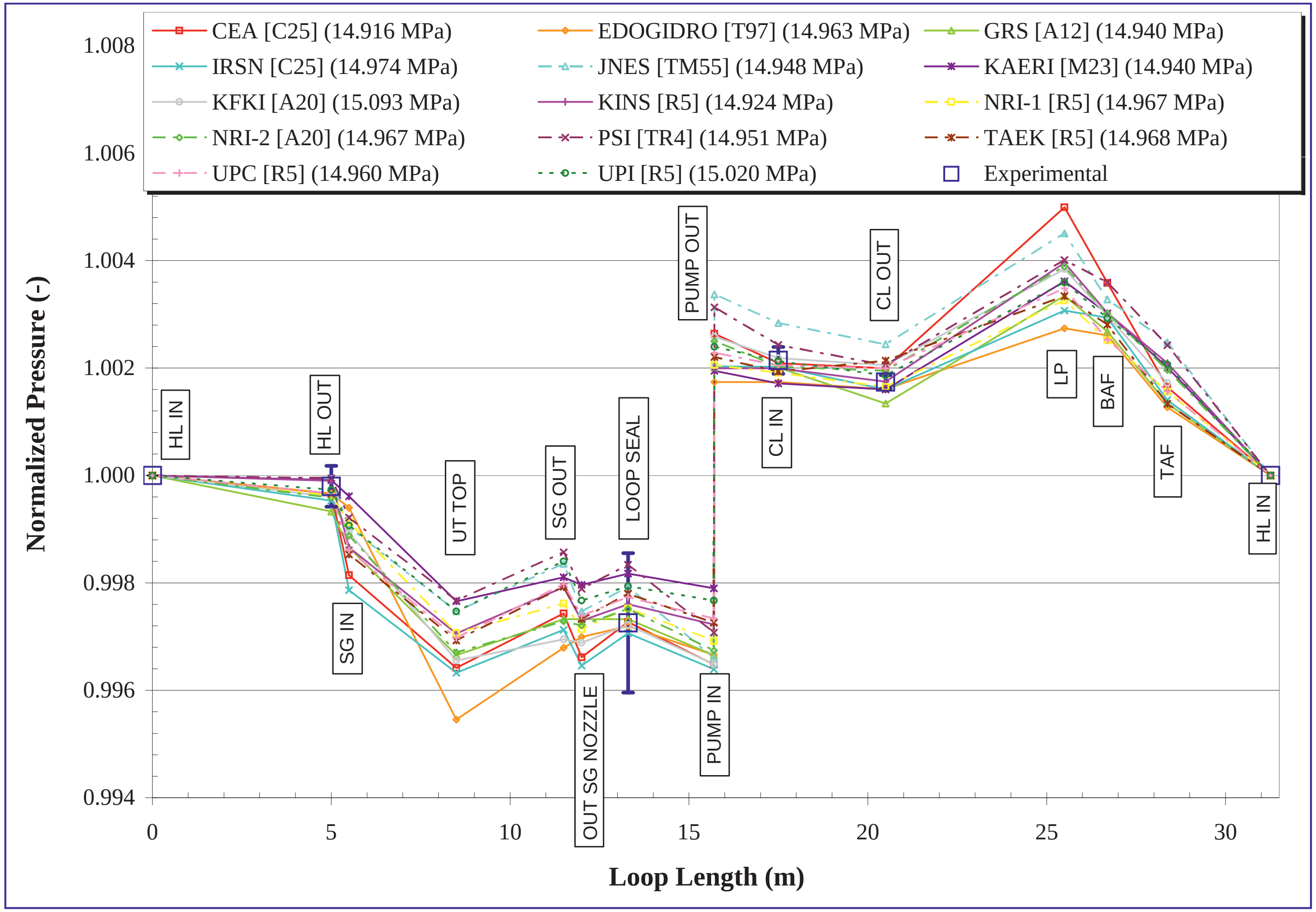

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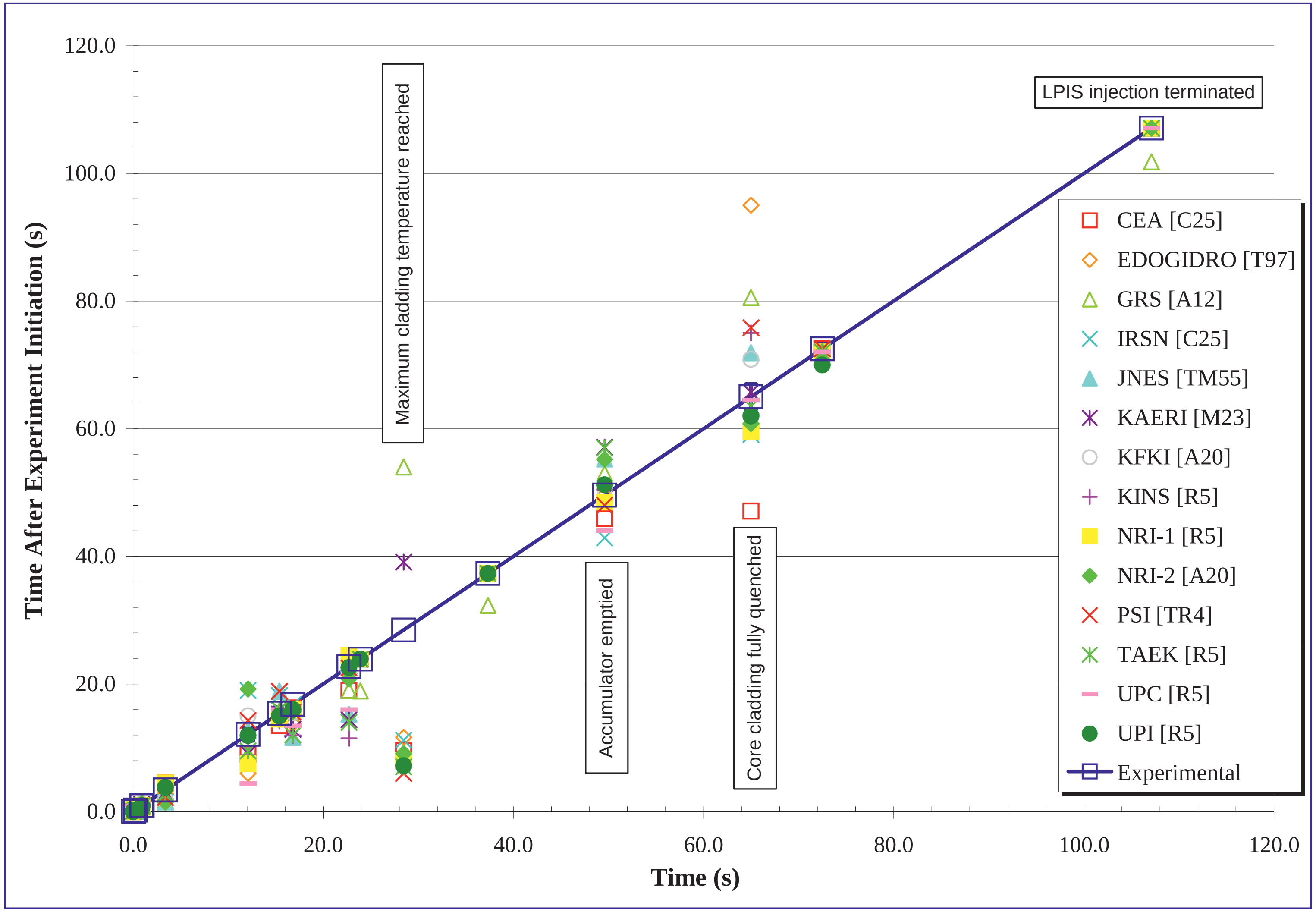
4. Selected results
4.1 Application to LOFT L2-5 experiment
Thermalhydraulic aspects



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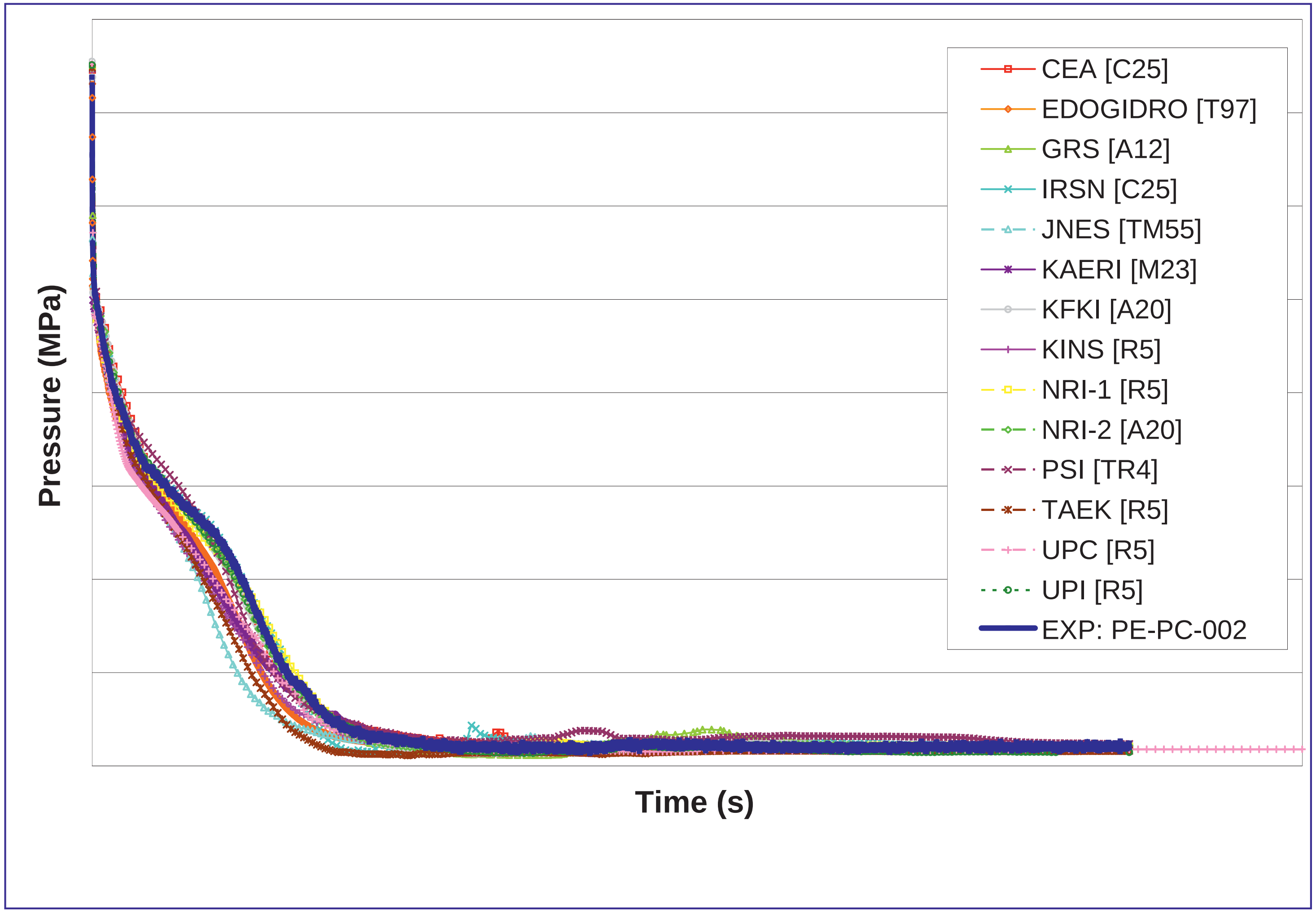
4. Selected results
4.1 Application to LOFT L2-5 experiment
Thermalhydraulic aspects



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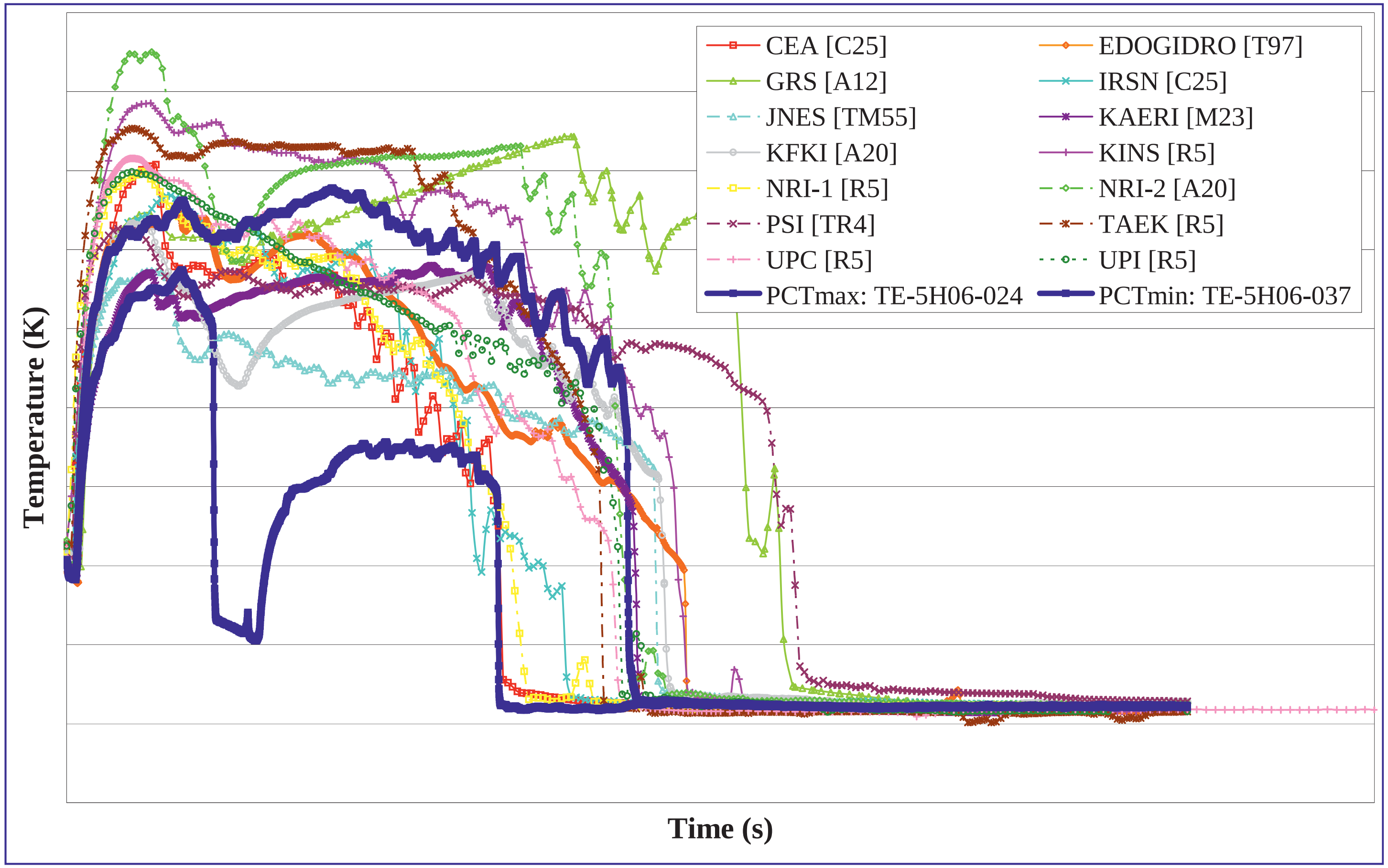
4. Selected results
4.1 Application to LOFT L2-5 experiment
Thermalhydraulic aspects



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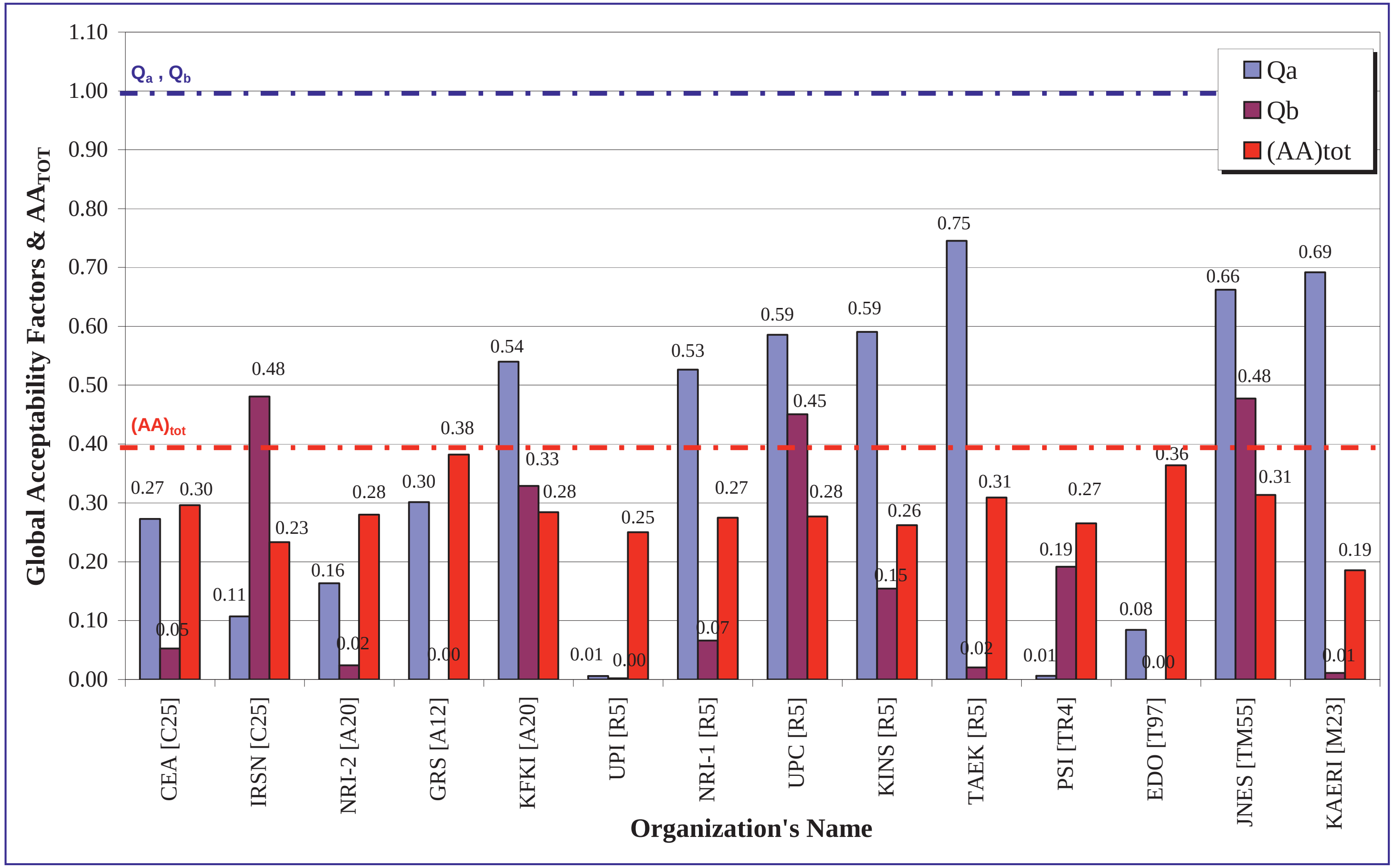
4. Selected results
4.1 Application to LOFT L2-5 experiment
Thermalhydraulic aspects

CODE and USER'S EFFECT ON BEMUSE Phase II



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4. Selected results
4.1 Application to LOFT L2-5 experiment
Thermalhydraulic aspects

conclusions

a) Almost all performed calculations appear qualified against the fixed criteria: few mismatches between results and acceptability thresholds have been characterized

b) Dispersion bands of results appear substantially less than in ISP-13: this testifies of code improvements in the last 20 years but especially in techniques for performing analysis.

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4. Selected results
4.1 Application to LOFT L2-5 experiment
Uncertainty aspects

OBJECTIVE: Estimation of the 5% and 95% percentiles with a confidence level of 0.95 for the 6 output parameters:

- Scalar output parameters:
  - First Peak Cladding Temperature ($MaxT_{clad}$ and $t < t_{inj}$)
  - Second Peak Cladding Temperature ($MaxT_{clad}$ and $t < t_{inj}$)
  - Time of accumulator injection
  - Time of complete quenching ($T_{clad} \leq T_{sat}$ + 30K)
- Time trends output parameters:
  - Maximum cladding temperature
  - Upper plenum pressure

4. Selected results
4.1 Application to LOFT L2-5 experiment
Uncertainty aspects

Percentile p: fraction p of a population that falls below that value.

Confidence level: a measure of how reliable a statistical result is.

Treated uncertain parameters:

- Physical models (e.g. Heat transfer correlations)
- Initial and boundary conditions (e.g. Initial total power)
- Material propertires (e.g. Fuel conductivity)
- Numerical parameters (e.g. Convergence criterion)
- Alternative models
- …

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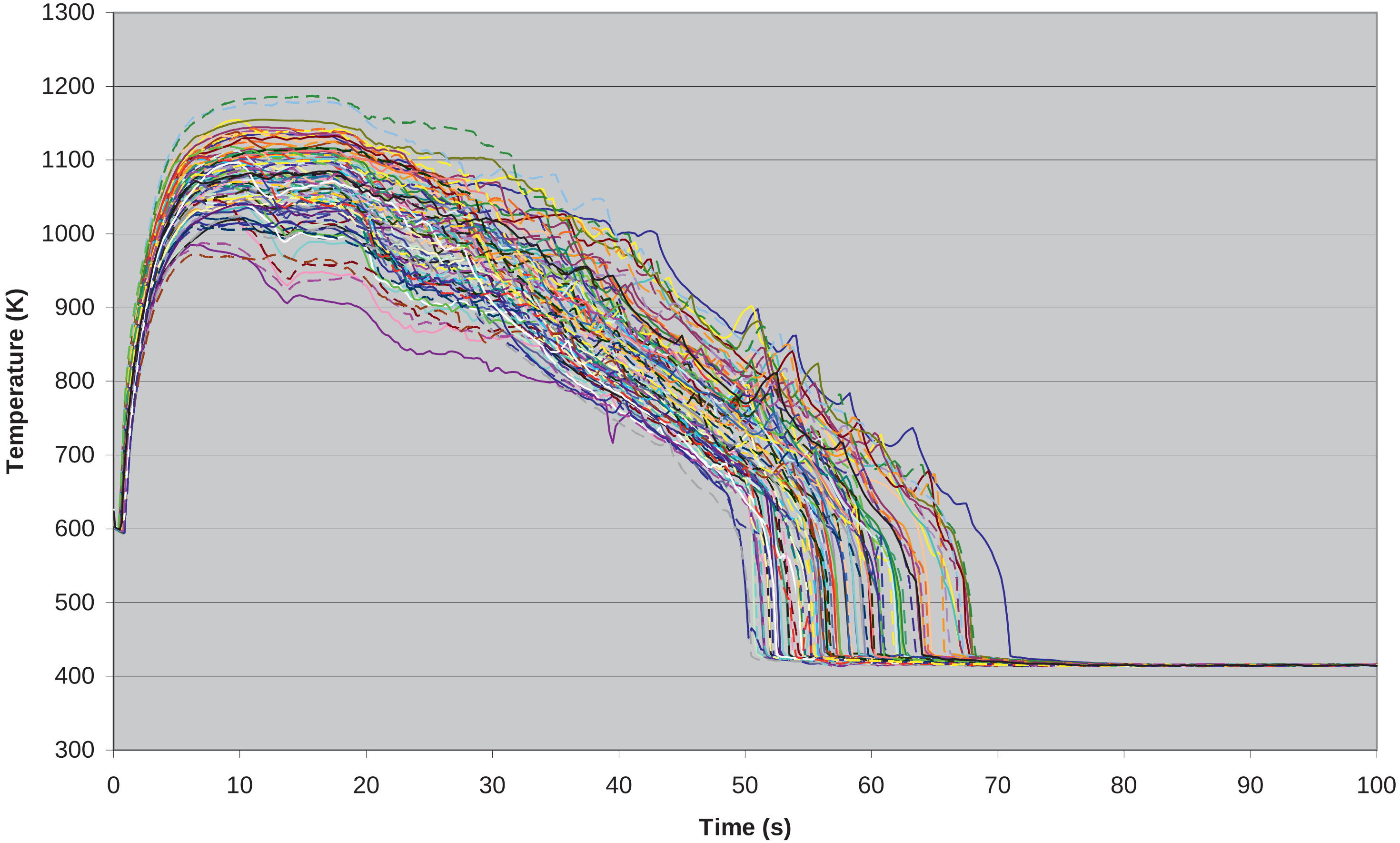
4. Selected results
4.1 Application to LOFT L2-5 experiment
Uncertainty aspects



Maximum cladding temperature. 100 calculations.

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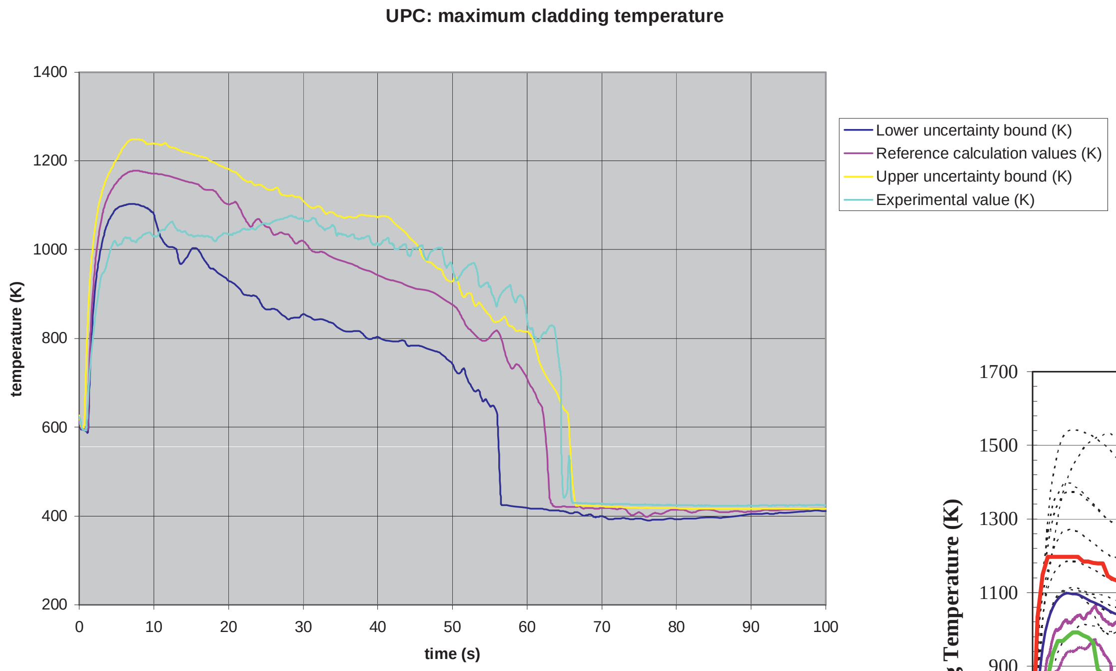
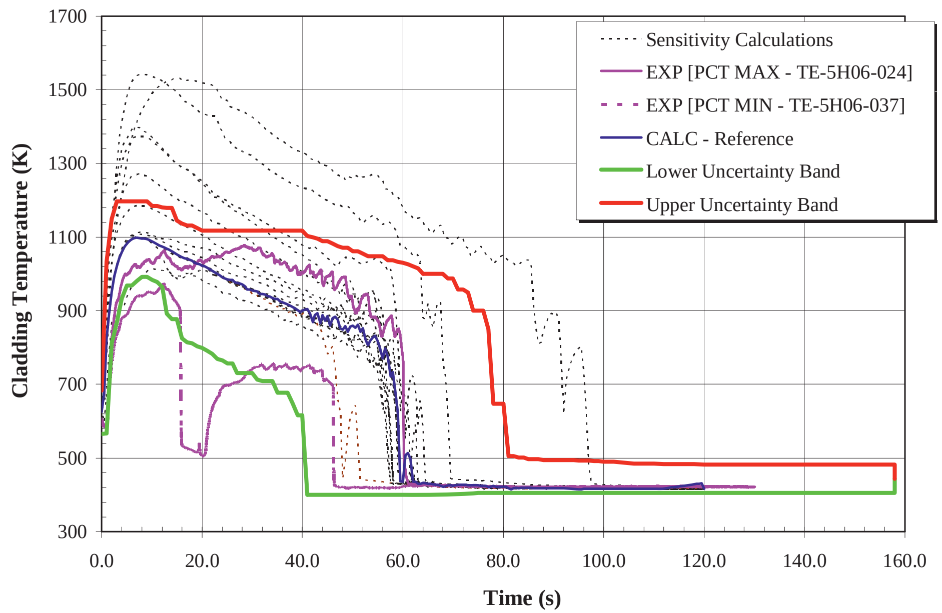
4. Selected results
4.1 Application to LOFT L2-5 experiment
Uncertainty aspects

Maximum cladding temperature - uncertainty bands.





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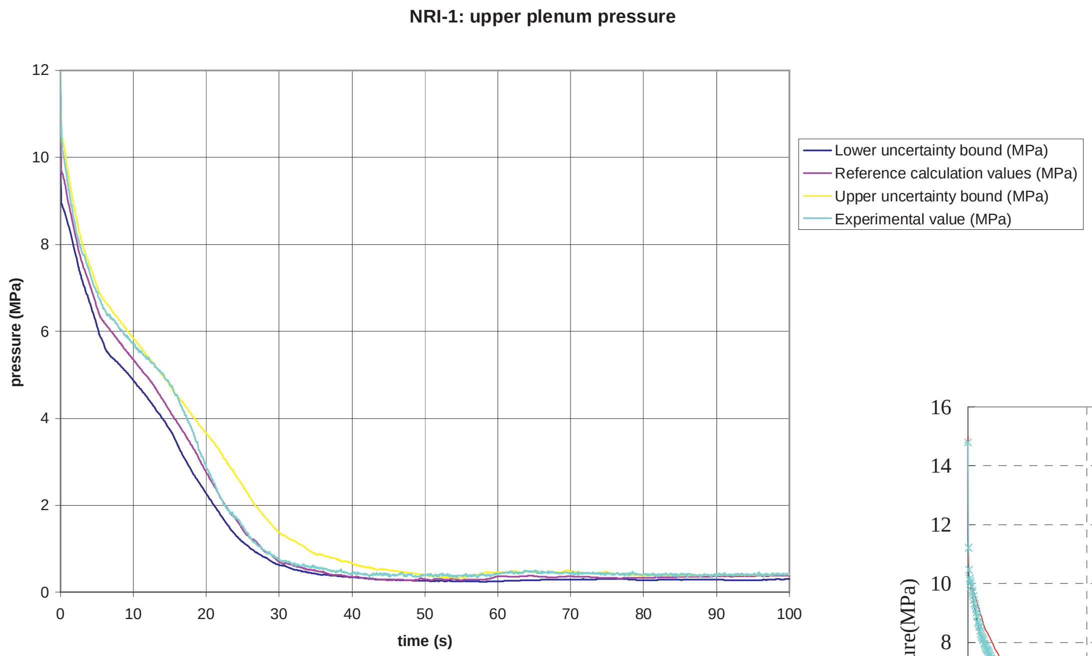
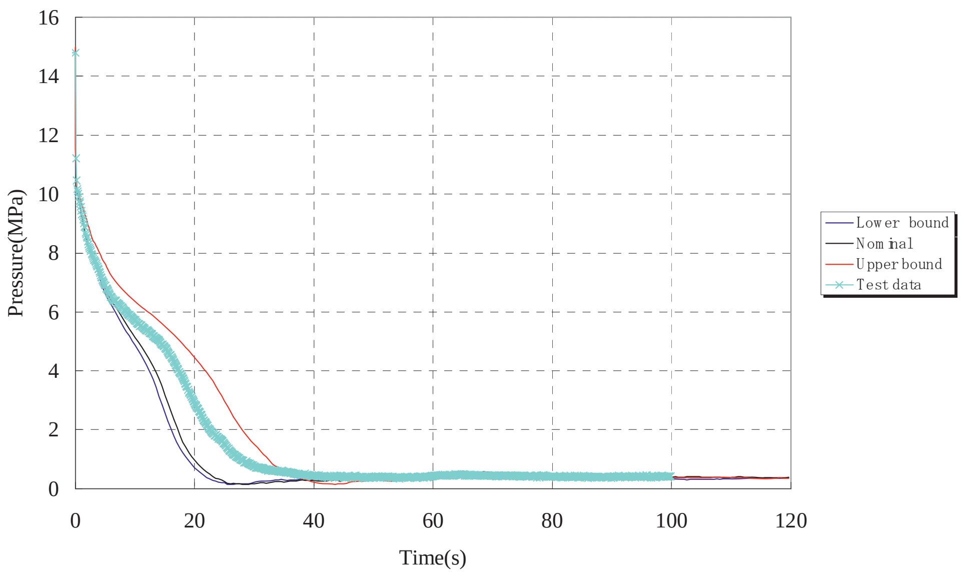
4. Selected results
4.1 Application to LOFT L2-5 experiment
Uncertainty aspects

Upper plenum pressure - uncertainty bands.





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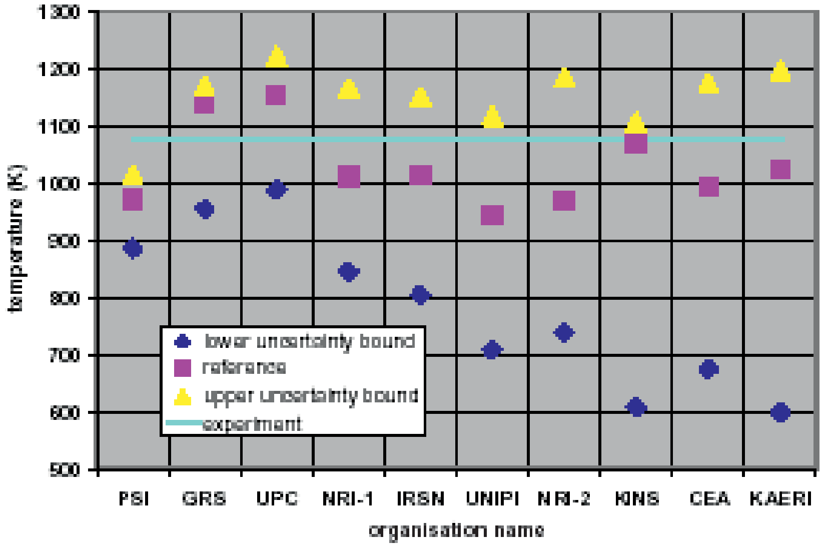
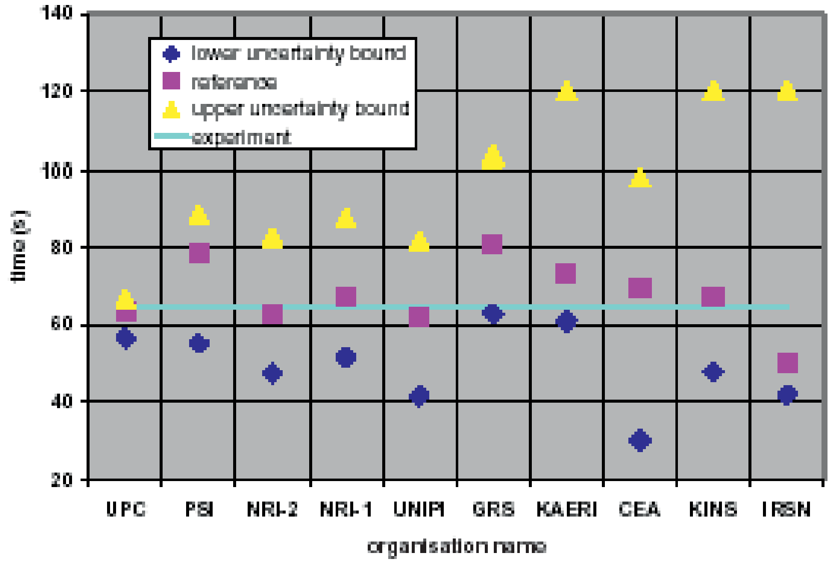
4. Selected results
4.1 Application to LOFT L2-5 experiment
Uncertainty aspects

2nd PCT: uncertainty bounds ranked by increasing band width



Time of complete quenching: uncertainty bounds ranked by increasing band width



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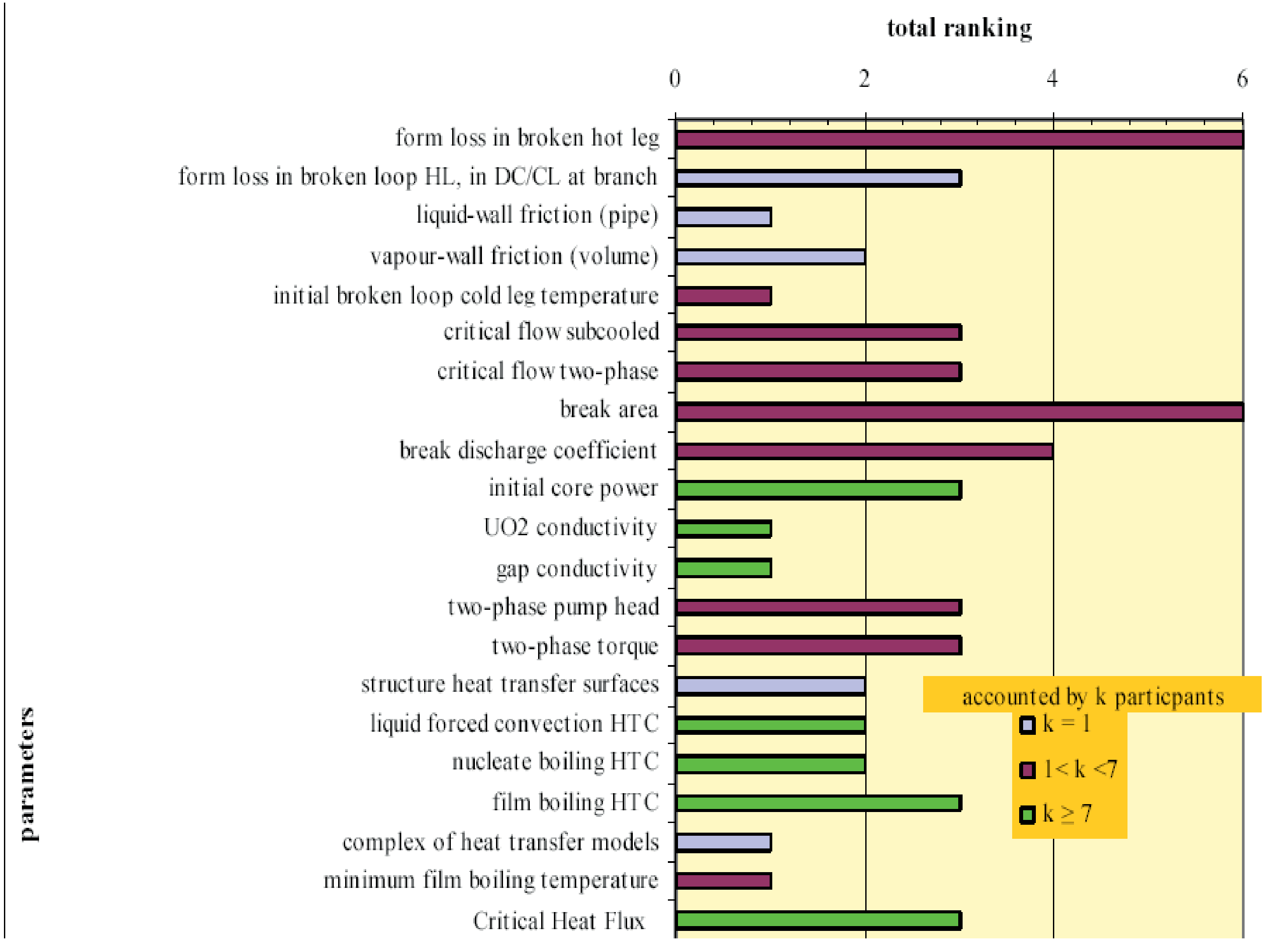
4. Selected results

4.1 Application to LOFT L2-5 experiment

Uncertainty aspects

Total ranking of the influence on the primary pressure per parameter



4. Selected results
4.1 Application to LOFT L2-5 experiment
Uncertainty aspects

Phase III recommendations

- Increasing number of code runs.
- Simple Random Sampling (SRS) when using Wilks’
- Failures treatment: correction / conservative approach: perform more code runs.
- Input uncertainty association: reduction of expert judgement by increasing experimental data base.

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4. Selected results
4.2 Application to Zion nuclear power plant
Thermalhydraulic aspects

Phase IV is the best-estimate analysis of an NPP-LBLOCA

The coordinator is Thecnical University of Catalonia (UPC)

The selected plant Zion (located nearby the city of Zion, Lake County, Illinois)

4 loop Pressurized Water Reactor

Westinghouse design / 3250 MWth

Date started: June 1973
Date closed: January 1998

4. Selected results
4.2 Application to Zion nuclear power plant
Thermalhydraulic aspects

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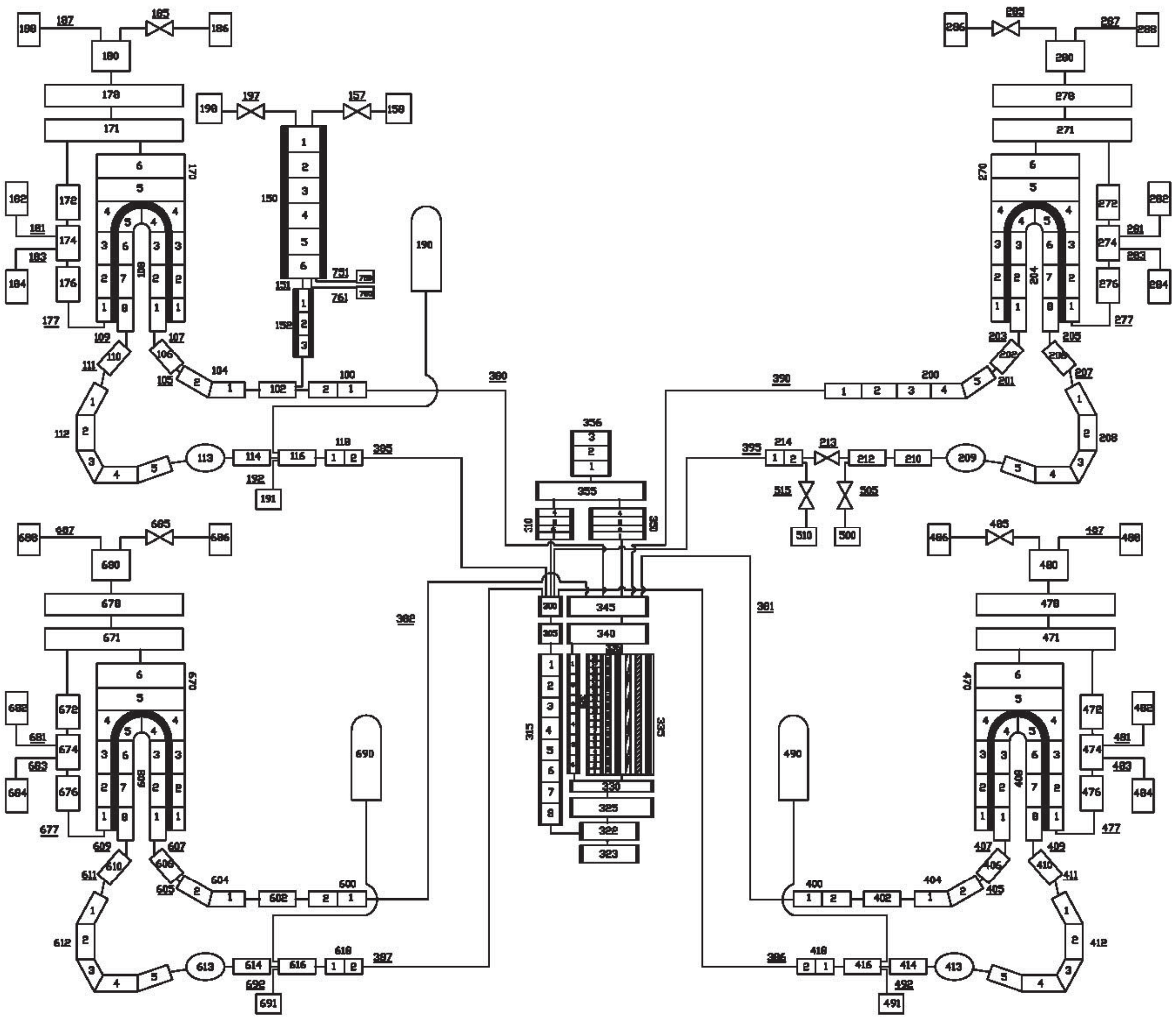

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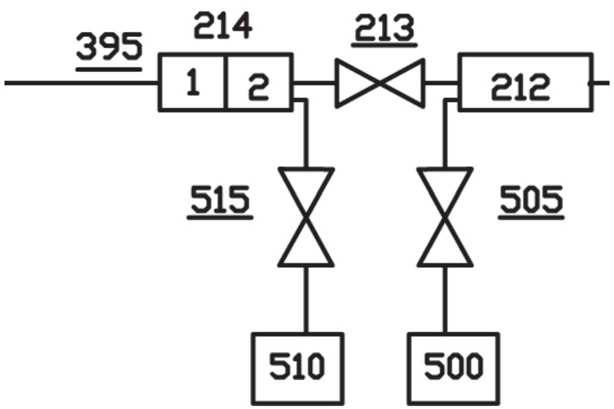
4. Selected results
4.2 Application to Zion nuclear power plant

Thermalhydraulic aspects



BREAK

Valves 515 and 505 :

- Full open area= 0.3832 m^2

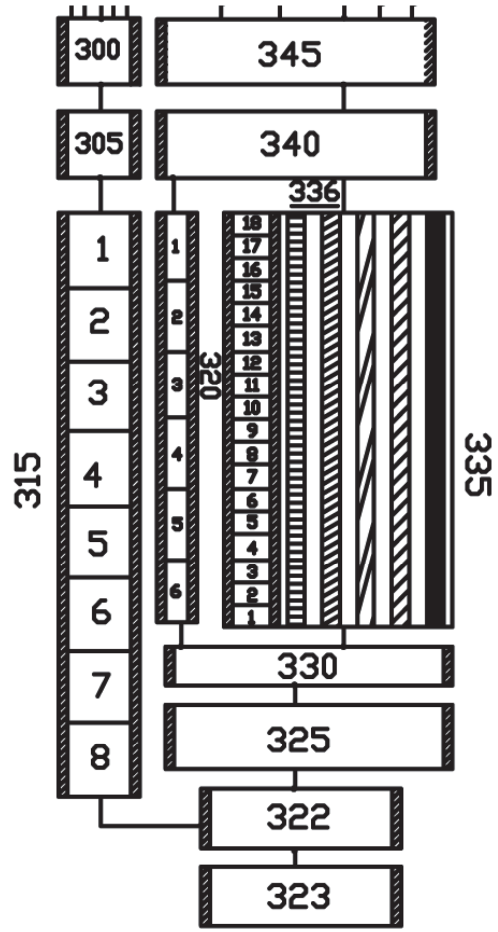
- Forward and reverse flow energy loss coefficients (Reynolds number independent), $A_F = A_R = 1$.

Volumes 510 and 500 simulate the pressure conditions of the containment.

4. Selected results
4.2 Application to Zion nuclear power plant

OECD/CSNI Workshop
ETSEIB-UPC
Barcelona November 2011

Thermalhydraulic aspects



- Core is nodalized with a 18 nodes pipe, pipe number 335
- Core fuel is simulated by 5 heat structures. Direct moderator heating is considered.
- Loss coefficients (forward and reverse) simulating grid spacers = 0.8077.
- Core bypass is nodalized with a 6 node pipe, pipe number 320.

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4. Selected results

4.2 Application to Zion nuclear power plant

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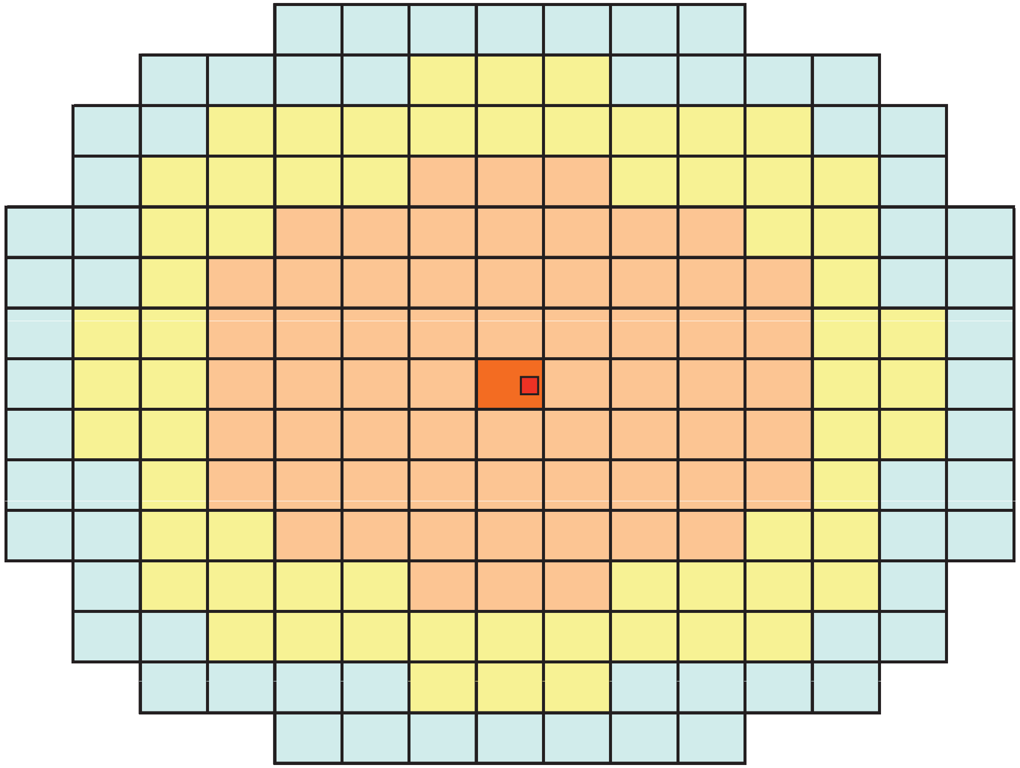
Thermalhydraulic aspects



#FA	rods per FA =	204	fuel rods
64	peripheral channel		13056
64	average channel		13056
64	hot channel		13056
1	hot FA in hot channel		203
1 rod	hot rod in hot FA		1
193	TOTAL		39372

4. Selected results
4.2 Application to Zion nuclear power plant

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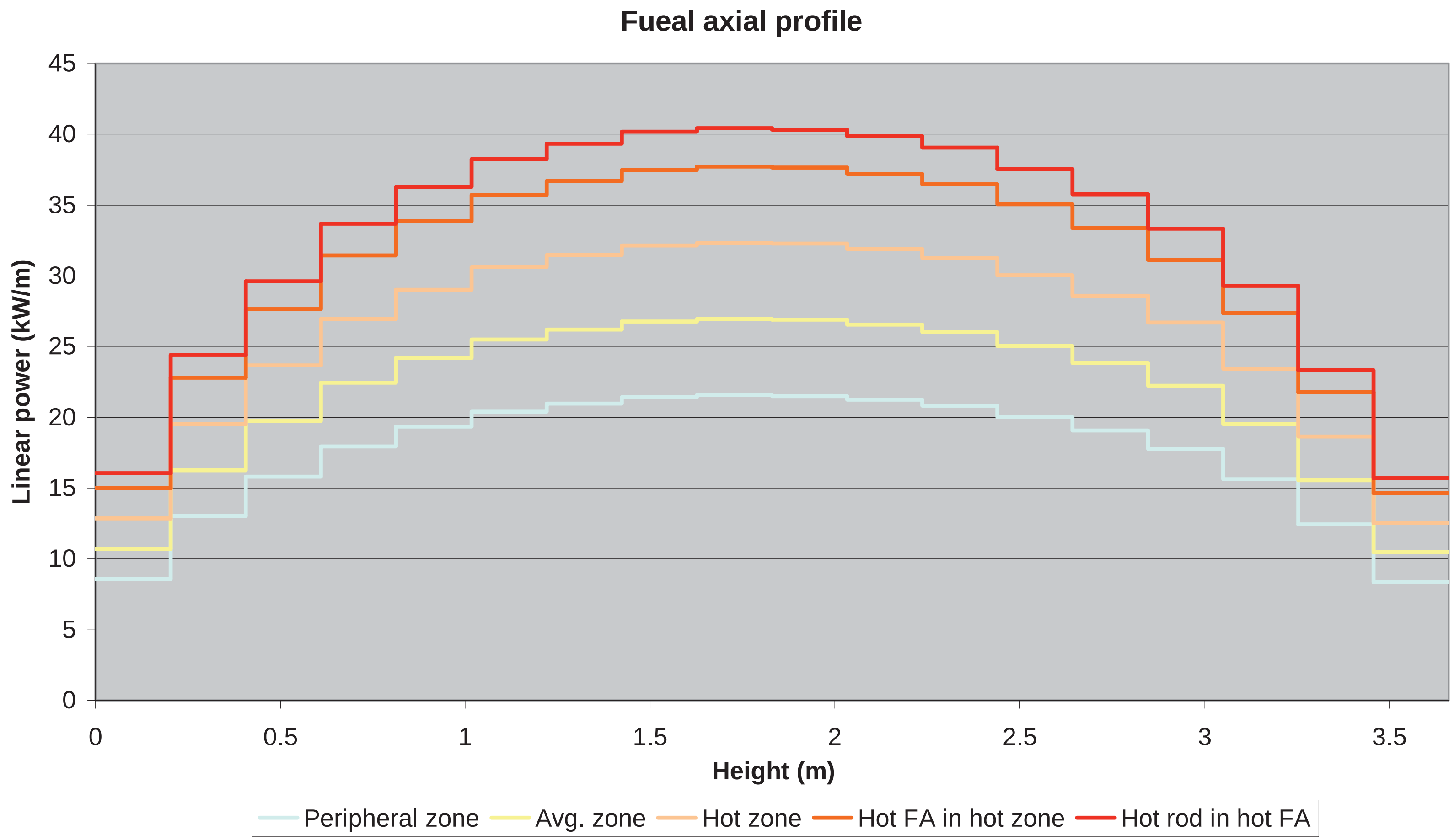
Barcelona November 2011

Thermalhydraulic aspects



4. Selected results

4.2 Application to Zion nuclear power plant

Thermalhydraulic aspects

Core Zones	Rod average linear power (kW/m)	Power per rod (kW/m)	Maximum linear power (kW/m)	Number of rods	Fuel power (kW)	Moderator power (kW)	Total power (MW)
Zone 1	17.56	64.25	21.56	13056	838881.02	21509.77	860.39
Zone 2	21.94	80.32	26.94	13056	1048601.27	26887.21	1075.49
Zone 3	26.33	96.38	32.33	13056	1258321.53	32264.65	1290.59
Zone 4	30.72	112.44	37.72	203	22825.71	585.27	23.41
Zone 5	32.92	120.47	40.42	1	120.47	3.09	0.12
Total				39372	3168750	81250	3250

193 Fuel Assemblies

204 rods per FA

Moderator = 2.5 %

Fuel = 97.5 %

Radial profile is assumed to be flat inside the fuel pellet.

4. Selected results
4.2 Application to Zion nuclear power plant

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Thermalhydraulic aspects

Parameter	Steady state value
Power (MW)	3250.0
Pressure in cold leg (MPa)	15.8
Pressure in hot leg (MPa)	15.5
Pressurizer level (m)	3.74
Core inlet temperature (K)	571.9
Core outlet temperature (K)	603.1
Primary coolant flow (kg/s)	17089
Secondary pressure (MPa)	6.74
SG downcomer level (m)	12.4
Feed water flow per loop (kg/s)	439.19
Accumulator pressure (MPa)	4.14
Accumulator gas volume (m^3)	14.83
Accumulator liquid volume (m^3)	23.39
RCPs velocity (rad/s)	120.06

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4. Selected results
4.2 Application to Zion nuclear power plant

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Thermalhydraulic aspects

Event	Time (s)
Break	0.0
SCRAM	0.0
Reactor coolant pumps trip	0.0
Steam line isolation	10.0
Feed water isolation	20.0
HPIS	NO

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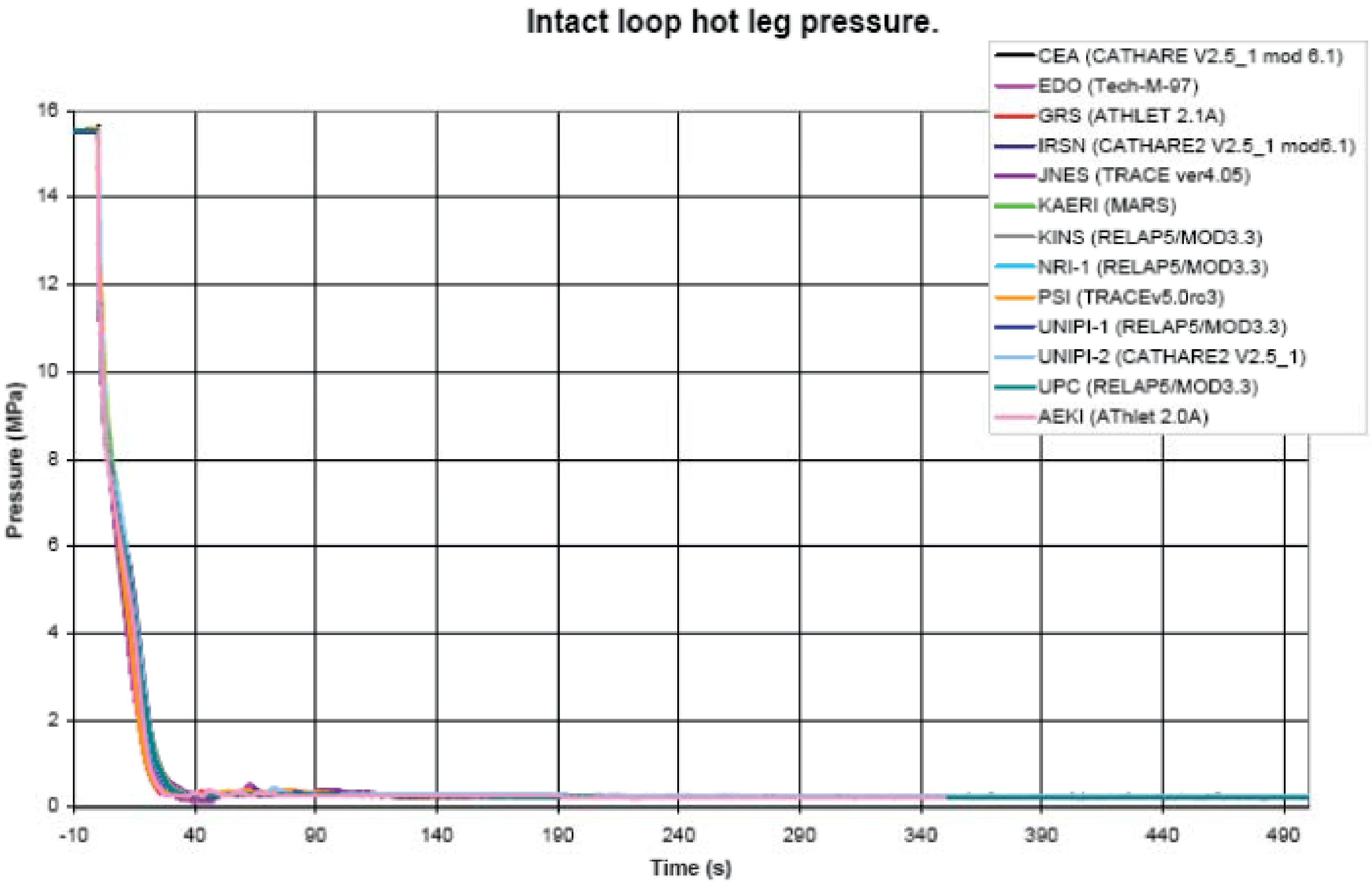
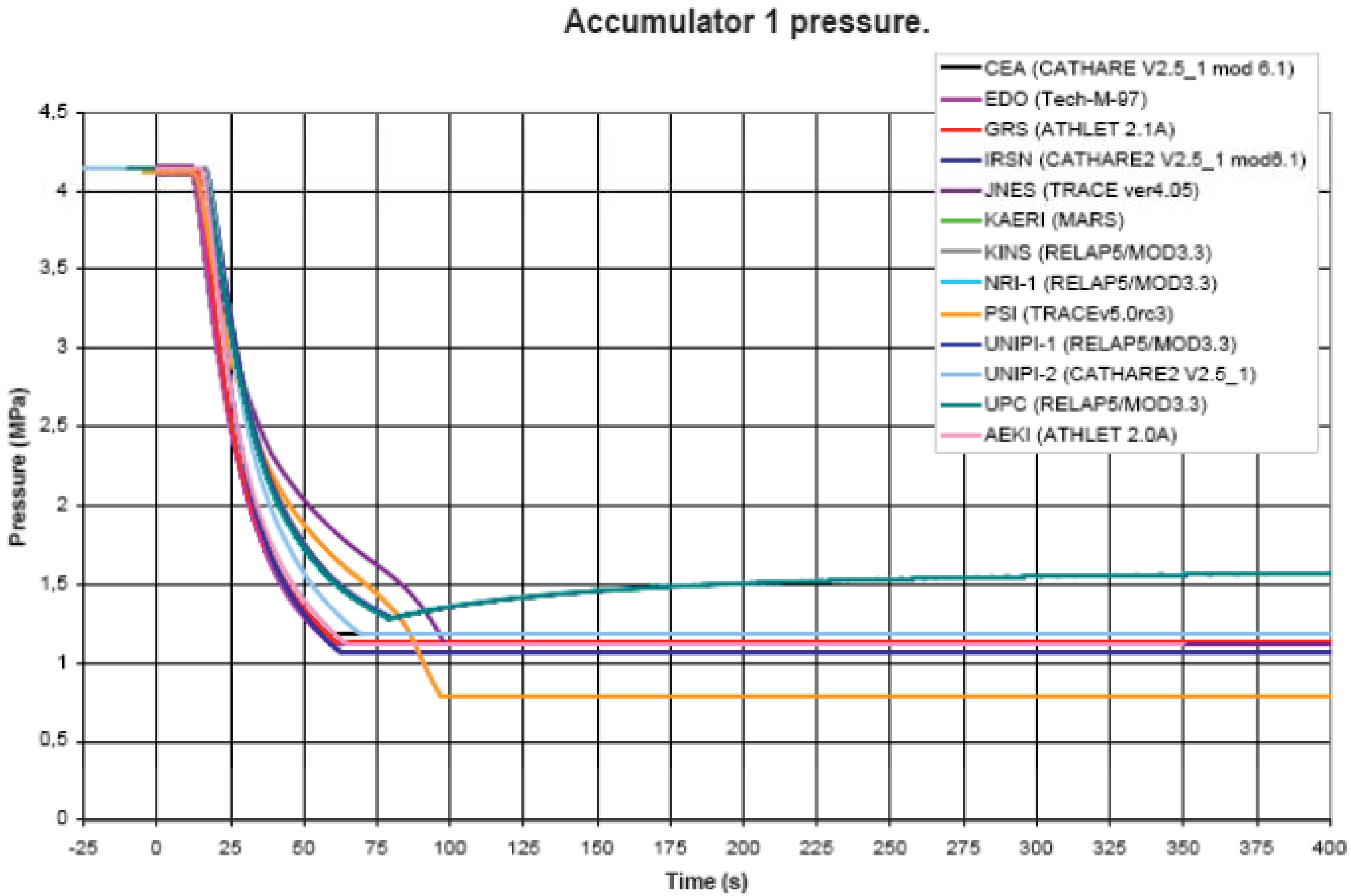
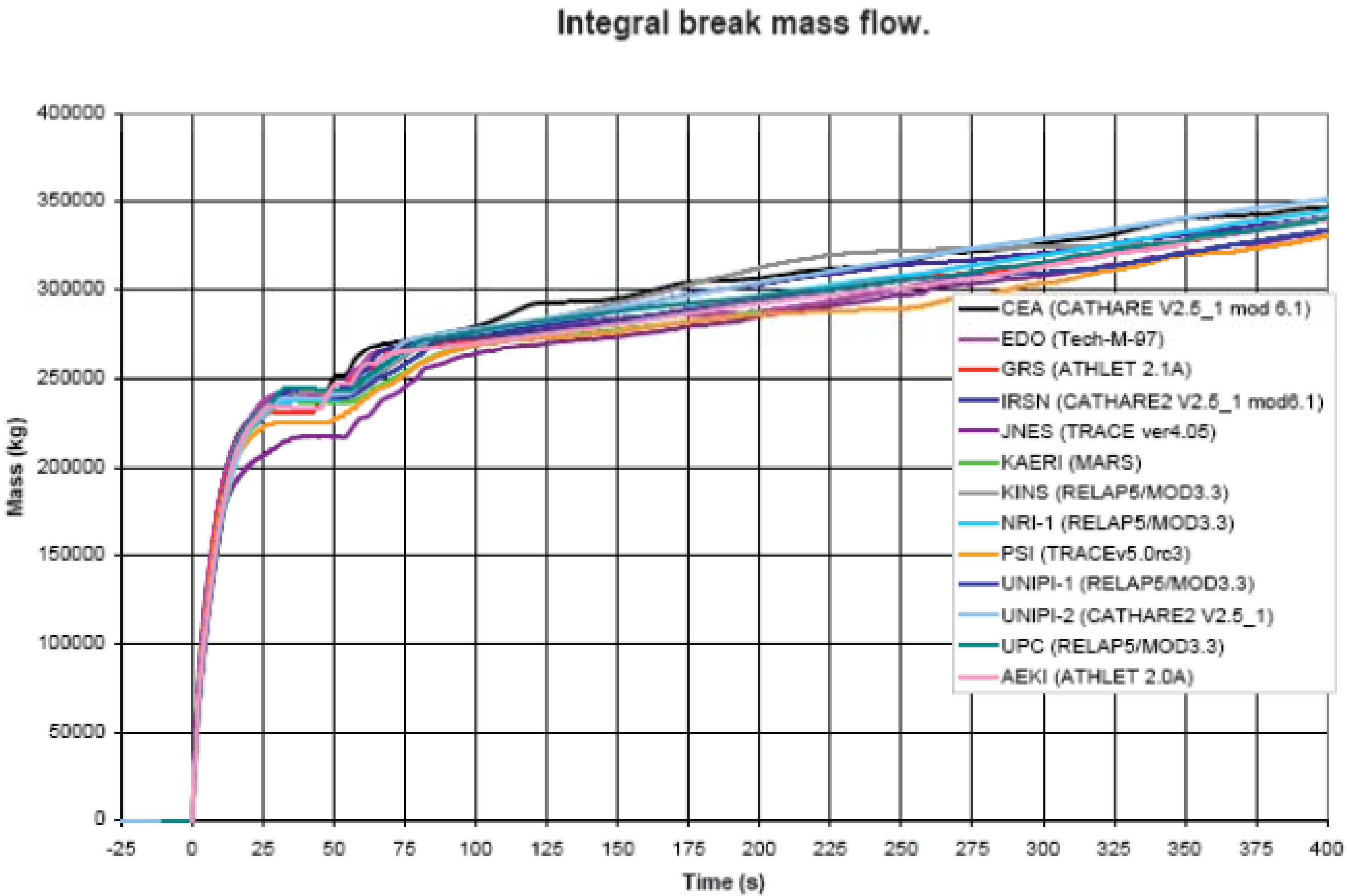
4. Selected results
4.2 Application to Zion nuclear power plant
Thermalhydraulic aspects



4. Selected results
4.2 Application to Zion nuclear power plant
Thermalhydraulic aspects

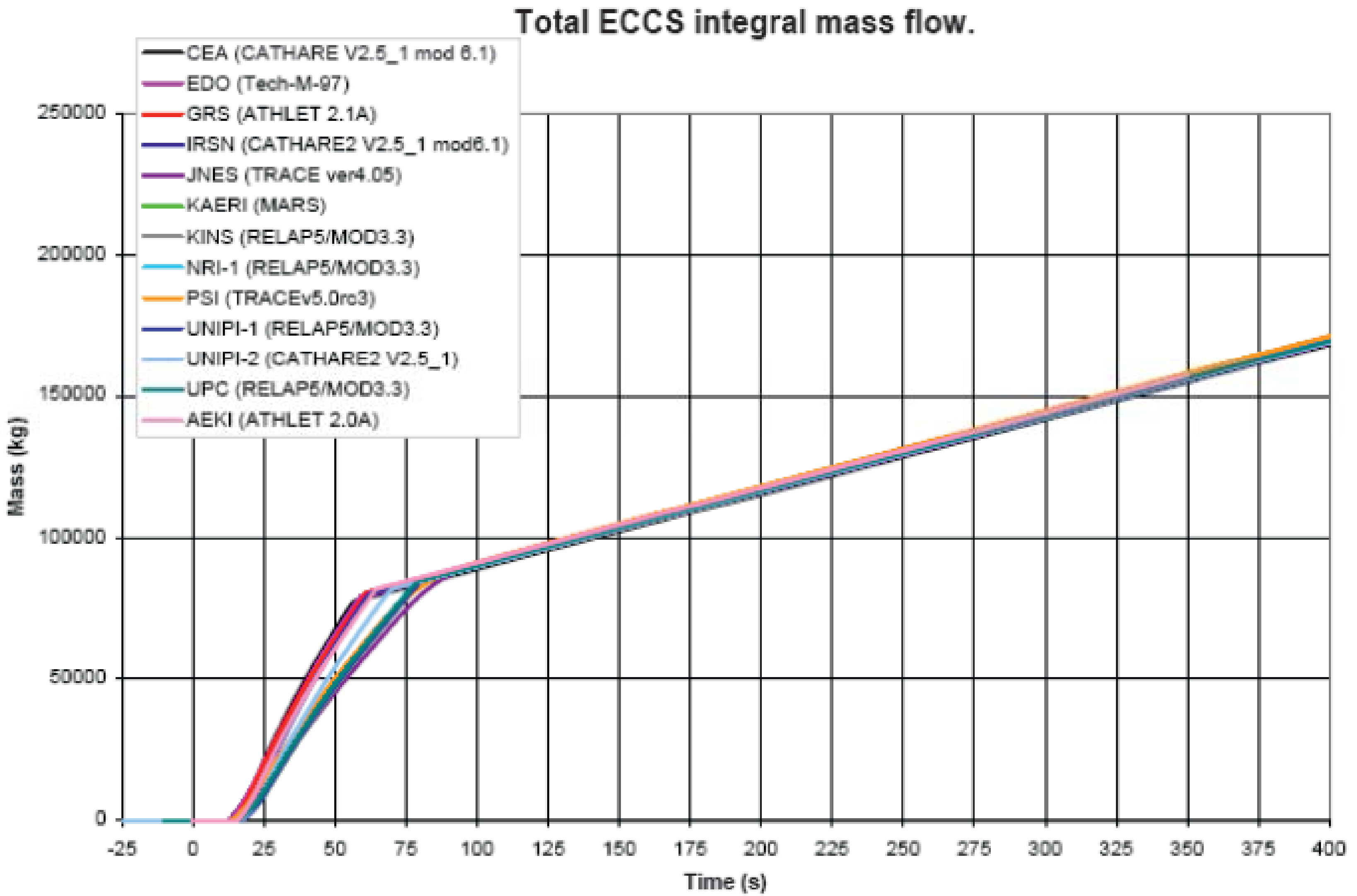


4. Selected results
4.2 Application to Zion nuclear power plant
Thermalhydraulic aspects



4. Selected results
4.2 Application to Zion nuclear power plant

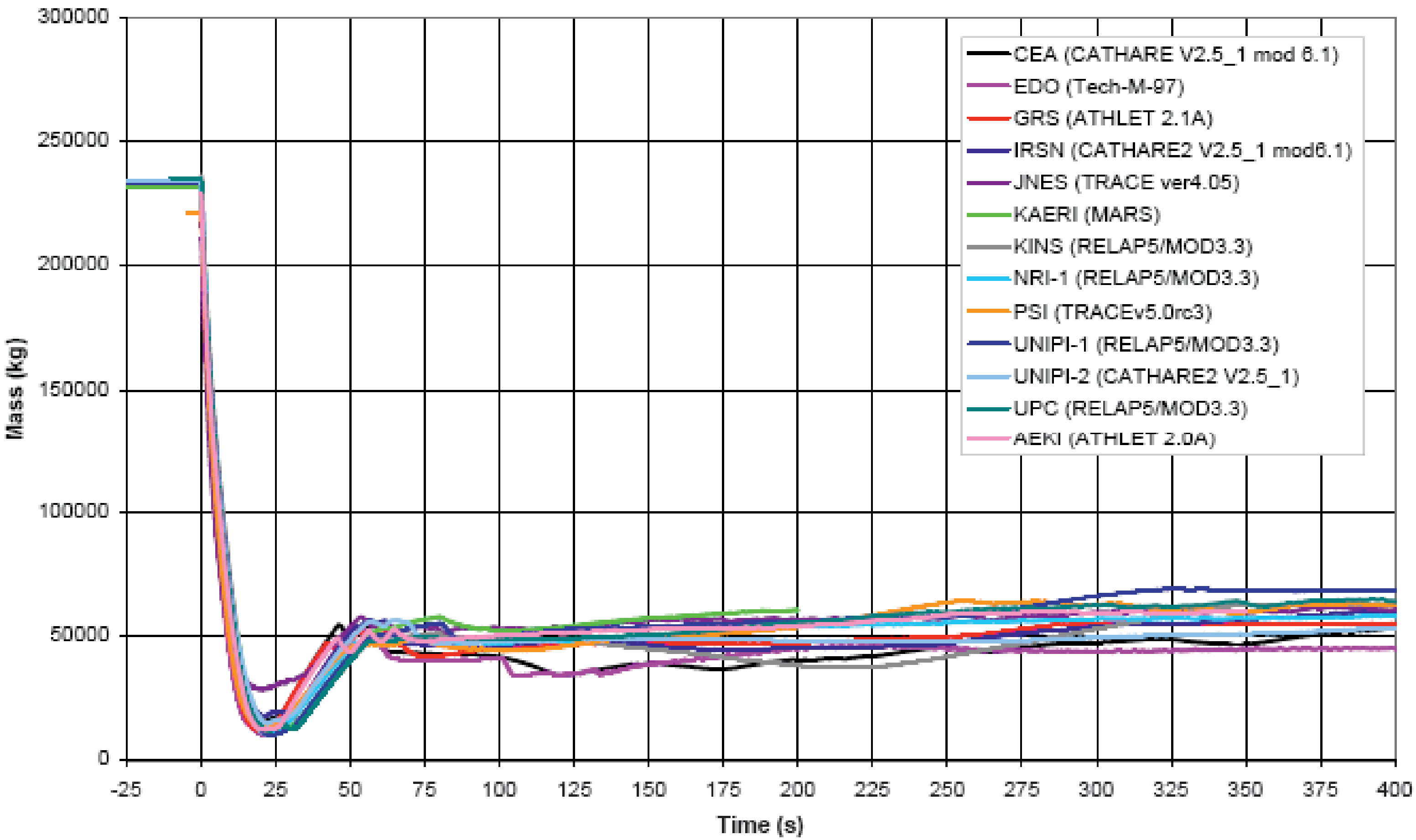
Thermalhydraulic aspects



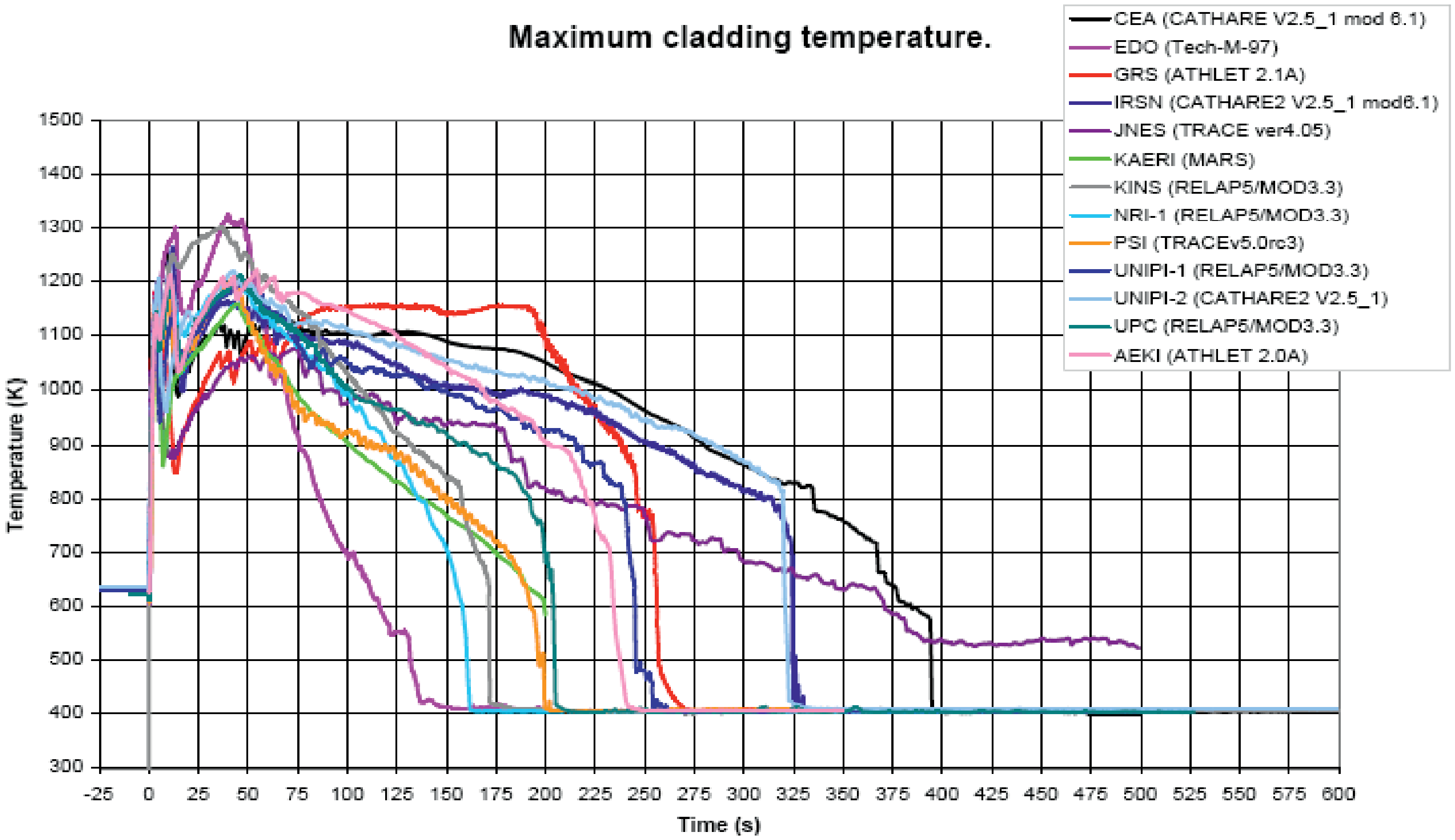
4. Selected results
4.2 Application to Zion nuclear power plant

Thermalhydraulic aspects

Primary side mass.



4. Selected results
4.2 Application to Zion nuclear power plant
Thermalhydraulic aspects



4. Selected results

4.2 Application to Zion nuclear power plant

Thermalhydraulic aspects

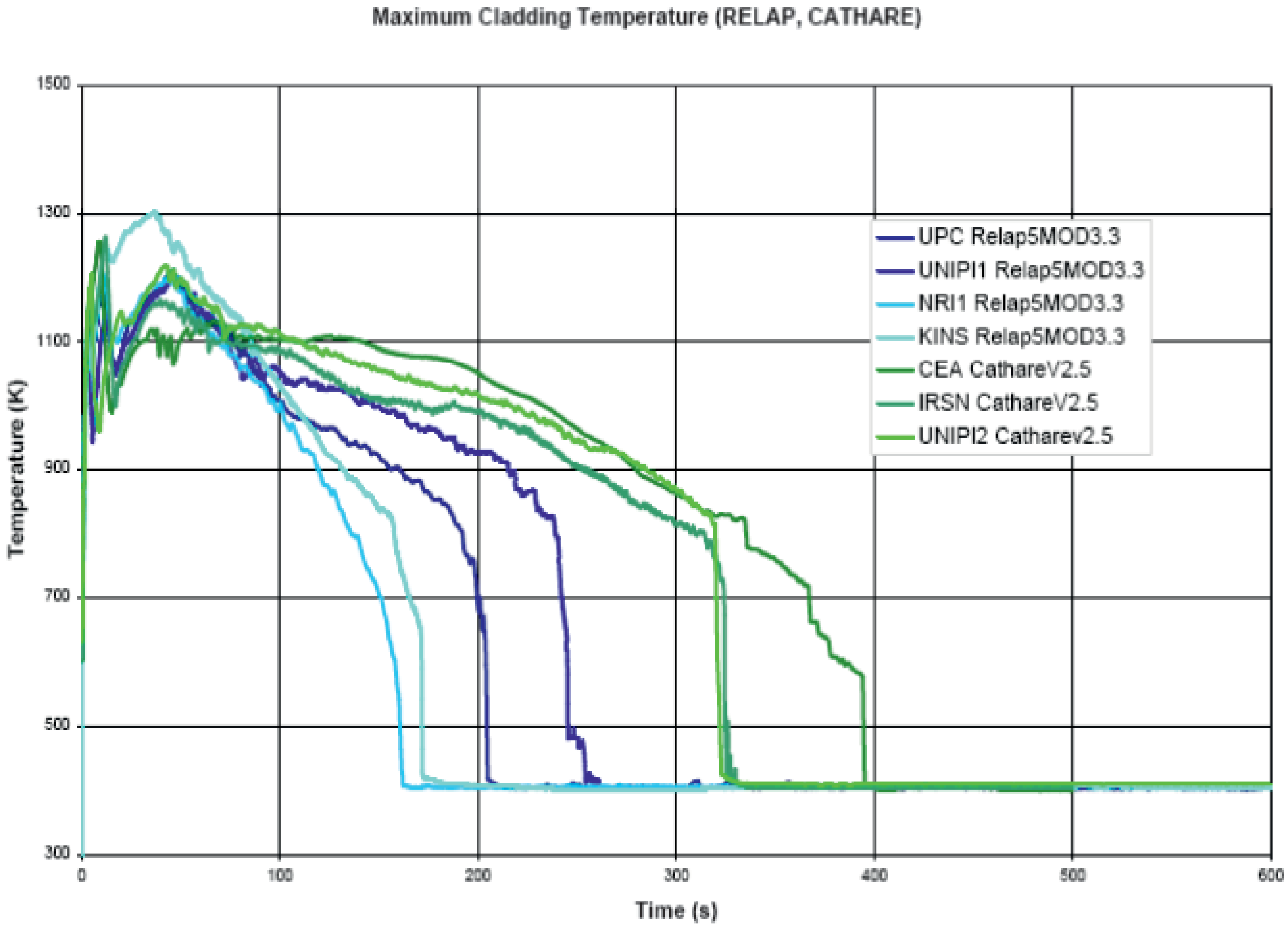
Despite of the dispersion shown in some of the figures, some events are predicted in a consistent way by participants among these:

Subcooled blowdown ended
Cladding temperature initially deviated from saturation (DNB in core)
Pressurizer emptied
Accumulator injection initiated
LPIS injection initiated

4. Selected results
4.2 Application to Zion nuclear power plant

Thermalhydraulic aspects

Code effect vs. User effect



4. Selected results
4.2 Application to Zion nuclear power plant
Thermalhydraulic aspects

All participants managed to simulate the scenario and predict the main parameters with credible consistency

Maximum values of PCT predicted are quite close one each other

PCT time trends and timing of complete core rewet still show some disagreements

A database, including comparative tables and plots has been produced. This database is suitable for providing the explanations needed for the following phases.

4. Selected results
4.2 Application to Zion nuclear power plant
Uncertainty and Sensitivity Analysis

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Common input parameters associated with a specific uncertainty, range of variation and type of probability density function (1 of 3)

Phenomenon	Parameter	Imposed range of variation	Type of pdf	Comments
Flow rate at the break	Containment pressure	[0.85, 1.15]	Uniform	Multiplier.
Fuel thermal behaviour	Initial core power	[0.98; 1.02]	Normal	Multiplier affecting both nominal power and the power after scram.
	Peaking factor (power of the hot rod)	[0.95; 1.05]	Normal	Multiplier.
	Hot gap size (whole core except hot rod)	[0.8; 1.2]	Normal	Multiplier. Includes uncertainty on gap and cladding conductivities.
	Hot gap size (hot rod)	[0.8; 1.2]	Normal	Multiplier. Includes uncertainty on gap and cladding conductivities.
	Power after scram	[0.92; 1.08]	Normal	Multiplier
	UO2 conductivity	[0.9, 1.1] (T_{fuel} <2000 K) [0.8,1.2] (T_{fuel} >2000 K)	Normal	Multiplier. Uncertainty depends on temperature.

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4. Selected results
4.2 Application to Zion nuclear power plant

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Uncertainty and Sensitivity Analysis

Common input parameters associated with a specific uncertainty, range of variation and type of probability density function (2 of 3)

Phenomenon	Parameter	Imposed range of variation	Type of pdf	Comments
Fuel thermal behaviour	UO2 specific heat	[0.98, 1.02] (T_{fuel} <1800 K) [0.87,1.13] (T_{fuel} >1800 K)	Normal	Multiplier. Uncertainty depends on temperature.
Pump behaviour	Rotation speed after break for intact loops	[0.98; 1..02]	Normal	Multiplier.
	Rotation speed after break for broken loop	[0.9; 1.1]	Normal	Multiplier.
Data related to injections	Initial accumulator pressure	[-0.2; +0.2] MPa	Normal	
	Friction form loss in the accumulator line	[0.5; 2.0]	Log-normal	Multiplier.
	Accumulators initial liquid temperature	[-10; +10] °C	Normal	
	Flow characteristic of LPIS	[0.95 ; 1.05]	Normal	Multiplier.

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4. Selected results
4.2 Application to Zion nuclear power plant
Uncertainty and Sensitivity Analysis

Common input parameters associated with a specific uncertainty, range of variation and type of probability density function (3 of 3)

Phenomenon	Parameter	Imposed range of variation	Type of pdf	Comments
Pressurizer	Initial level	[-10; +10] cm	Normal	
	Initial pressure	[-0.1; +0.1] MPa	Normal	
	Friction form loss in the surge line	[0.5; 2]	Log-normal	Multiplier.
Initial conditions: primary system	Initial intact loop mass flow rate	[0.96; 1.04]	Normal	Multiplier. This parameter can be changed through the pump speed or through pressure losses in the system.
	Initial intact loop cold leg temperature	[-2; +2] K	Normal	This parameter can be changed through the secondary pressure, heat transfer coefficient or area in the U-tubes.
	Initial upper-head mean temperature	[T_{cold} ; T_{cold} + 10 K]	Uniform	This parameter refers to the "mean temperature" of the volumes of the upper plenum.

4. Selected results

4.2 Application to Zion nuclear power plant

Uncertainty and Sensitivity Analysis

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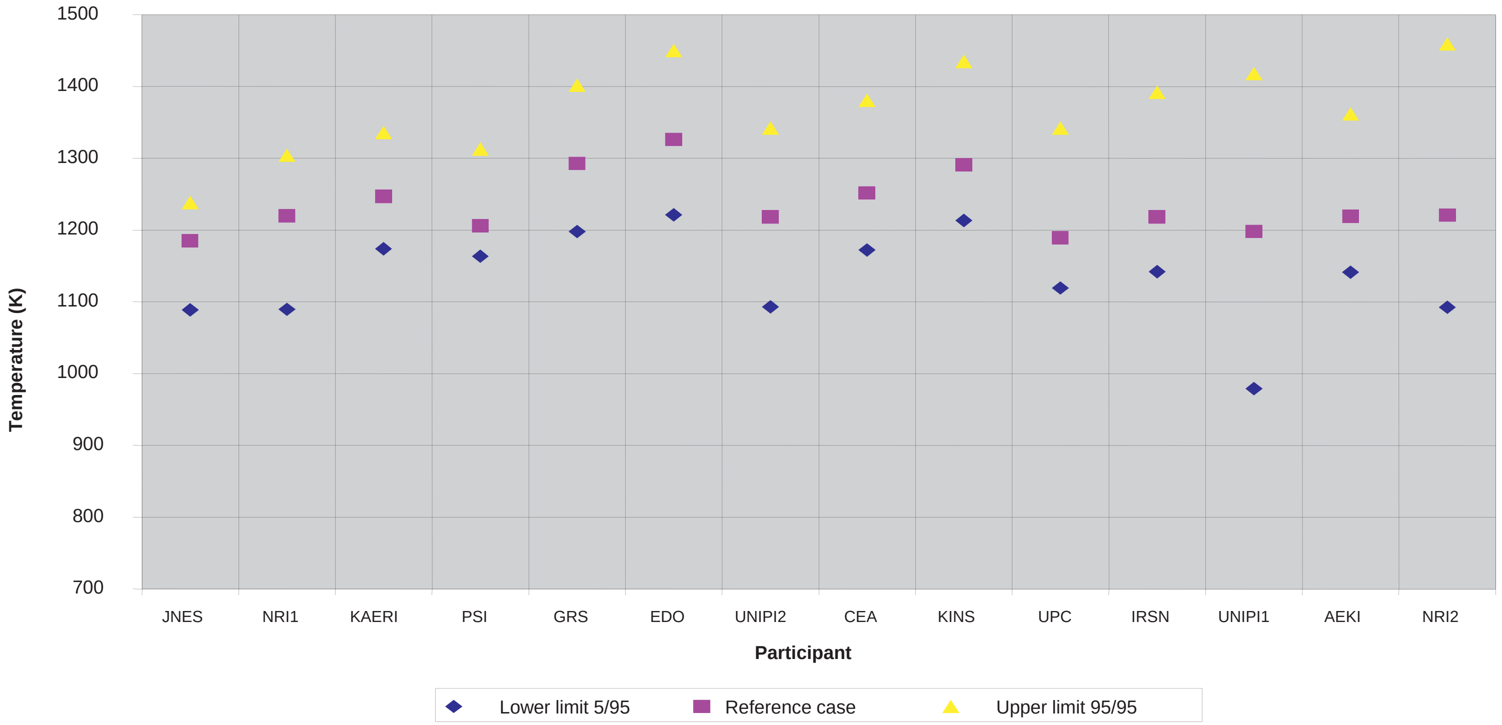
Calculated uncertainty bands of the maximum PCT of Zion NPP LB-LOCA



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4. Selected results
4.2 Application to Zion nuclear power plant
Uncertainty and Sensitivity Analysis

Comparing results for the maximum PCT, there is an overlap region of, roughly, 15K (between 1221K and 1238K). This region is very small

When not including participants with extreme values of the uncertainty bands, it is possible to obtain a better overlap region

Similar comment can be made if upper bound values of different participants are compared

4. Selected results
4.2 Application to Zion nuclear power plant

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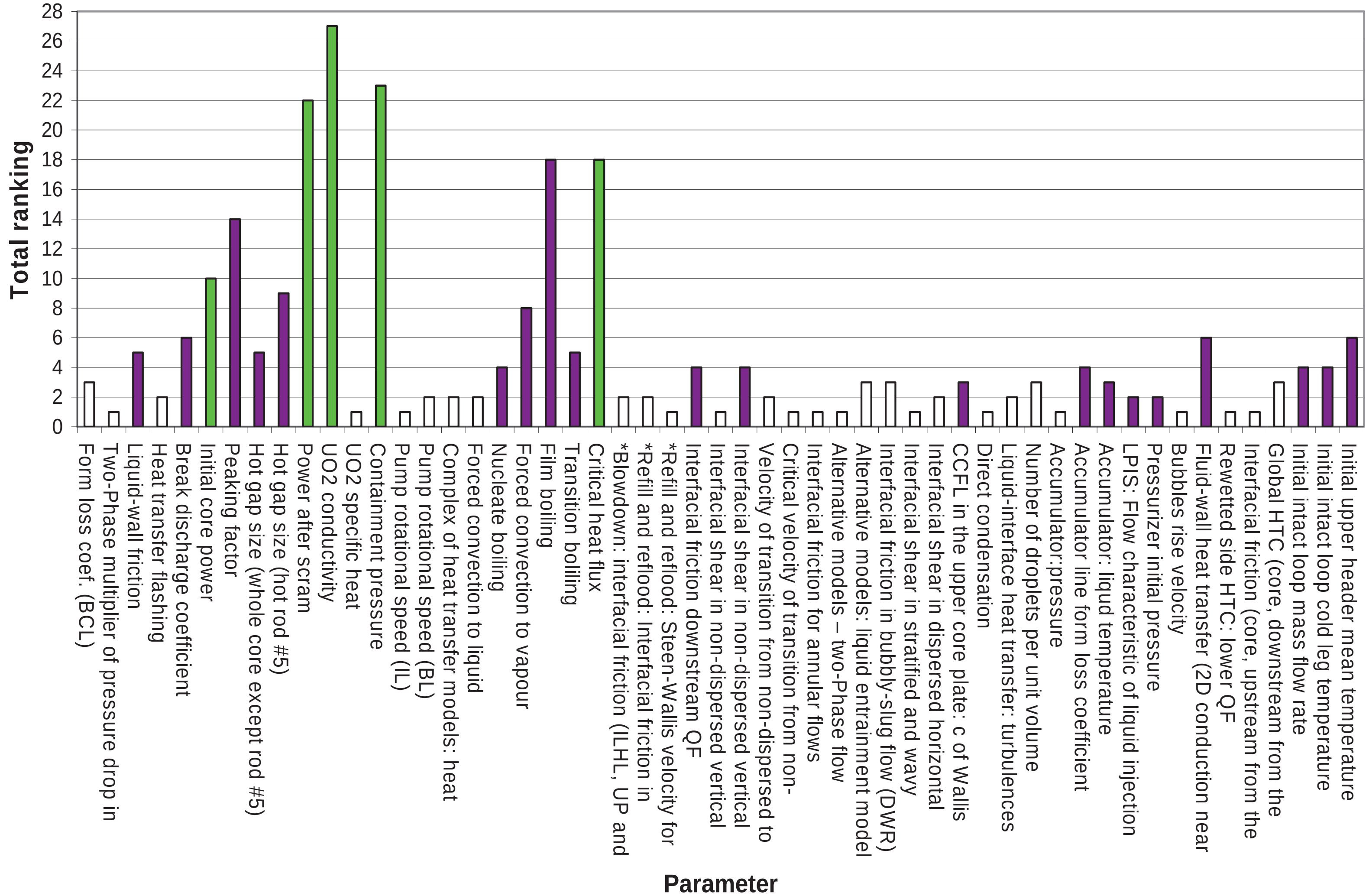
Uncertainty and Sensitivity Analysis



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Summary

1. Objectives of the programme
2. Main steps
3. Used methods
4. Selected results
   - 4.1 Application to LOFT L2-5 experiment
   - 4.2 Application to Zion nuclear power plant
5. Conclusions and recommendations

5. Conclusions and recommendations

The methods used are considered to be mature for application, including licensing processes

Differences in the application of the methods and in the results are observed. The project is a step forward in identifying and solving them

User effect can also be seen in applications of uncertainty methods

Recommendation for the proper use of Wilks' formula have been produced and shared by participants

BEMUSE brought some evidence that more effort and maybe specific procedures should be focused on the determination of input uncertainties.

This last point is an issue for recommendation for further work

Discussion of OECD LWR Uncertainty Analysis in Modeling (UAM) Benchmark

K. Ivanov, M. Aramova, E. Royer, J. Gulliford

CSNI Workshop on
OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations

Barcelona, Spain, 16-18 November 2011
School of Industrial Engineering of Barcelona

Outline

- Introduction
- OECD LWR UAM Benchmark Activity
- Status and Results of Phase I (Neutronics Phase)
- Status of Phase II (Core Phase)
- Priorities of Phase III (System Phase)
- Conclusions and Outlook

Introduction

- **The principles that support the risk-informed regulation* should be considered in an integrated decision-making process**
- **Thus, any evaluation of licensing issues supported by a safety analysis should take into account both deterministic and probabilistic aspects of the problem**
- **The deterministic aspects should be addressed using best estimate coupled code calculations and considering the associated uncertainties**
- **In recent years there has been an increasing demand from nuclear research, industry, safety and regulation for best estimate predictions to be provided with their confidence bounds**

* Incorporating an assessment of safety significance or relative risk in NRC regulatory actions. Making sure that the regulatory burden imposed by individual regulations or processes is commensurate with the importance of that regulation or process to protecting public health and safety and the environment.

Introduction

- Uncertainty Analysis in Modeling (UAM) Expert Group (EGUAM) – focuses on benchmark activities, which contribute to establishing a unified framework to estimate safety margins, which would provide more realistic, complete and logical measure of reactor safety:
  - Completed LWR coupled code benchmarks: LWR Core Transient Benchmarks, TMI PWR MSLB, PB-2 BWR TT, Ringhals BWR Stability, PWR MOX REA, and Kozloduy VVER-1000 CT
  - On-going LWR coupled code benchmarks – Kalinin-3 VVER-1000 and Oskarshamn-2 BWR Stability
- OECD LWR UAM benchmark - uncertainty propagation is being estimated through the whole simulation process on a unified benchmark framework to provide credible coupled code predictions with defensible uncertainty estimations of safety margins at the full core/system level
- Objective - the chain of uncertainty propagation from basic data, and engineering uncertainties, across different levels (multi-level), and physics phenomena (multi-physics) to be tested on a number of benchmark exercises for which experimental data is available and for which the power plant details have been released

Safety Implications

Among the expected results of this project are:

- **Systematic consideration of uncertainty and sensitivity methods in all steps. This approach will generate a new level of accuracy and will improve transparency of complex dependencies**
- **Systematic identification of uncertainty sources**
- **All results will be represented by reference results and variances and suitable tolerance limits**
- **The dominant parameters will be identified for all physical processes**
- **Support of the quantification of safety margins**
- **The experiences of validation will be explicitly and quantitatively documented**
- **Recommendations and guidelines for the application of the new methodologies will be established**

Expected Results

Among the expected results of this project are:

- **Experience on sensitivity and uncertainty analysis (several methods will be used and compared):**
  - **Deterministic methods**
  - **Statistical sampling techniques**
  - **Hybrid methodologies**
- **New developments:**
  - **Adapted Global Sensitivity Analysis:**
    - **Allows for combination of different input sources of uncertainties**
  - **Non-linear ESM approach to Hessian Matrix Construction:**
    - **In neutronics calculations, the responses behave linearly within cross-sections variations, so first-order approximation are acceptable.**
    - **In multi-physics uncertainty quantification, thermal-hydraulics feedback is expected to be strong, thereby higher order approximations will be needed.**

Safety Implications

MOTIVATION		OBJECTIVE
First order SA is well-established and widely used but…..	➡	**Second Order Approximation** (Quadratic Prediction)
n inputs ⇒ n(n+1)/2 unknowns in Hessian matrix ⇒ n(n+1)/2 equations ⇒ n(n+1)/2 code runs ⇒ **Too Expensive**	➡	**Reduction of Computational Cost** in **Hessian Matrix Construction** (minimize required code runs)

Source - NCSU

Safety Implications

- The OECD LWR UAM activity will establish an internationally accepted benchmark framework to compare, assess and further develop different uncertainty analysis methods associated with the design, operation and safety of LWRs

- As a result the LWR UAM benchmark will help to address current nuclear power generation industry and regulation needs and issues related to practical implementation of risk informed regulation

- The use of coupled codes supplemented with uncertainty analysis allows to avoid unnecessary penalties due to incoherent approximations in the traditional decoupled calculations, and to obtain more accurate evaluation of margins regarding licensing limit

- This becomes important for licensing power upgrades, improved fuel assembly and control rod designs, higher burn-up and others issues related to operating LWRs as well as to the new Generation 3+ designs being licensed now (ESBWR, AP-1000, EPR-1600 and etc.)

EGUAM

WPRS

Expert Group on Reactor Fuel Performance (EGRFP)	Expert Group on Radiation Transport and Shielding (EGRTS)	Expert Group on Reactor Physics and Advanced Nuclear Systems (EGRPANS)	Expert Group on Uncertainty Analysis in Modelling (EGUAM)

- Tomasz Kozlowski (KTH, Sweden) is leading EG
- Key activity is coordination of LWR transient benchmarks & workshops, led by Kostadin Ivanov, M. Avramova and K. Velkov
  - V1000-CT, BFBT, PSBT, KALININ, Oskarshamn
  - results of these will form part of multi-stage project for estimating uncertainties for coupled neutronics/thermal hydraulics modelling (UAM)

Special thanks to Kevin Hesketh, who is leading the WPRS, and Tomasz Kozlowski for their support of the OECD LWR UAM benchmark activity

Benefits of Participation

- To develop, propose and/or validate advanced uncertainty and sensitivity methodology
- Have access to different techniques in sensitivity / uncertainty analysis
- Compare and exchange of know-how, resolve difficulties with the world experts
- Improve understanding of model validity and their limitation
- Provide evidence to model simplification
- Have access to high quality integral experiments from experimental facilities and operating power reactors
- Acquire competence in quantifying confidence bounds for physics and safety parameters in best estimate methods required for licensing

Description

Phase I (Neutronics Phase)

Exercise I-1: "Cell Physics"

Exercise I-2: "Lattice Physics"

Exercise I-3: "Core Physics"

Phase II (Core Phase)

Exercise II-1: "Fuel Physics"

Exercise II-2: "Time-Dependent Neutronics"

Exercise II-3: "Bundle Thermal-Hydraulics"

Phase III (System Phase)

Exercise III-1: "Core Multi-Physics"

Exercise III-2: "System Thermal-Hydraulics"

Exercise III-3: "Coupled Core-System"

Exercise III-4: "Comparison of BEPU vs. Conservative Calculations"

UAM - LWR calculation

- I. Neutronics – Specification on Phase I
- II. Core – Specification on Phase II
- III. System – Specification on Phase III

Description

The OECD LWR UAM benchmark is organized following the modular approach, which allows for a maximum flexibility and benefit to the participants:

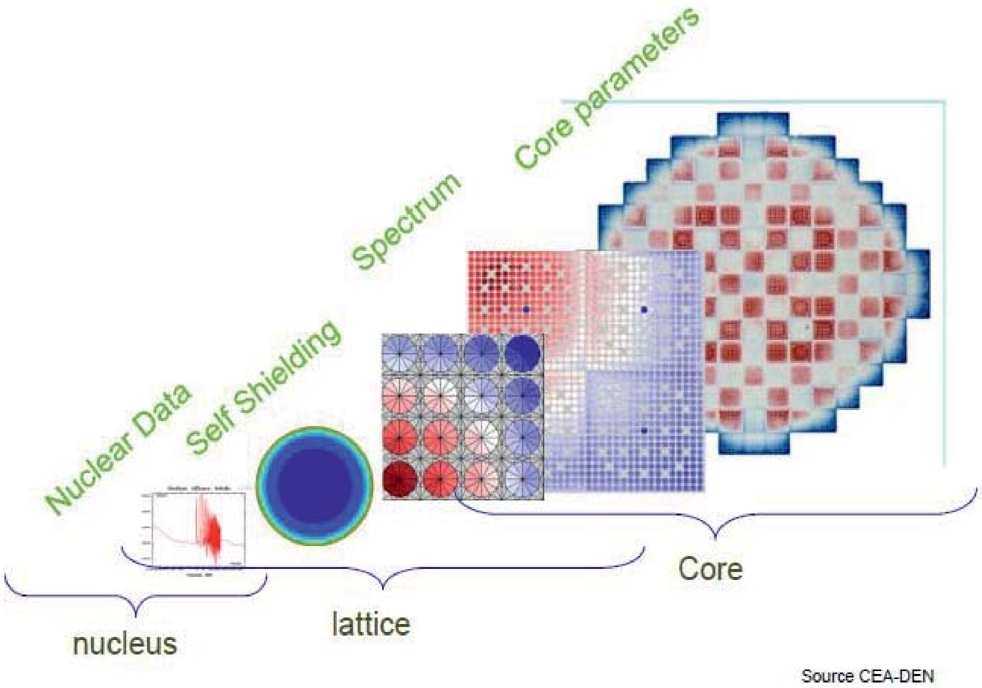
- **Participants can participate in the 3 Phases and in all exercises propagating the uncertainties through all stages of coupled reactor physics/thermal hydraulics calculation scheme**
- **Alternatively participants can participate in selected exercises (the benchmark team will provide them with the input data following the established format of previous OECD/NEA benchmarks) and just follow the activities in the other exercises**
- **There are several types of operating LWRs to be followed in this benchmark representative of a BWR, a PWR and a VVER. Participants can model one or more reactor types depending on their interests**
- **For each Exercise two types of test problems are designed – numerical test problems (provided with reference solutions) for these types of LWR reactors) and experimental test problems obtained from publicly available databases**

Description

- The OECD LWR UAM benchmark framework is based on the introduction of 10 steps (Exercises), which are grouped in 3 Phases
- For each exercise Input (I), Output (O), and propagated Uncertainty (U) parameters are identified
- Identifying the sources of Input (I) uncertainties for each Exercise - which input uncertainties are propagated from the previous exercise and which one are new?
- Other important parameters to be defined are the Output (O) uncertainties and propagated Uncertainty parameters (U) for each Exercise:
  - ✓ This task is directly related to the objective of each Exercise
  - ✓ The Output (O) uncertainties are for specified output parameters for each Exercise, used to test (evaluate) the utilized uncertainty method
  - ✓ The propagated Uncertainty parameters (U) are output parameters, which are selected to be propagated further through the follow-up Exercises in order to calculate the overall resulting uncertainty

Phase I

Phase I – Neutronics Phase:



- ✓ **Exercise 1 (I-1) – Cell Physics:** Derivation of the multi-group microscopic cross-section libraries

  U-1 (multi-group cross-section variance / covariance matrix)

- ✓ **Exercise 2 (I-2) – Lattice Physics:** Derivation of the few-group macroscopic cross-section libraries

  U-2 (two-group parameter variance / covariance matrix)

- ✓ **Exercise 3 (I-3) – Core Physics:** Criticality (steady state) stand-alone neutronics calculations

  U-3 (uncertainties in k_{eff}, power peaking factors, rod worth)

Phase I

- **Phase I is focused on understanding uncertainties in prediction of key reactor core parameters associated with LWR stand-alone neutronics core simulation**
- **Such uncertainties occur due to input data uncertainties, modelling errors, and numerical approximations**
- **Several LWR types are selected, based on previous benchmark experiences and available data, in order to address all industrial issues and participants interests.**
  - ✓ ***Representative of operating PWR based on Three Mile Island 1 (TMI-1) NPP***
  - ✓ ***Representative of operating BWR based on Peach Bottom-2 (PB-2) NPP***
  - ✓ ***Representative of operating VVER-1000 based on Kozloduy-6 and Kalinin-3 (V1000) NPPs***
  - ✓ ***Representative of Generation III PWRs with UOX and MOX cores***
- **The intention is to follow the established calculation scheme for LWR design and safety analysis in the nuclear power generation industry and regulation**

Phase I

- The chosen approach in Phase I is to select/propagate for each exercise the most important contributors which can be treated in a practical manner
- The cross-section uncertainty information is considered as the most important source of input uncertainty for Phase I
- The cross-section related uncertainties are propagated through the 3 Exercises of Phase I
- In Exercise I-2 manufacturing and geometry (technological) uncertainties are added to account for them in lattice physics calculations

Manufacturing uncertainties for TMI-1

Source - PSU

Parameter	Distribution	Mean	Sigma
Fuel density	Normal	10.283 g/cm^3	0.05666666 g/cm^3
Fuel pellet diameter	Normal	0.9391 cm	0.000433333 cm
Gap thickness	Normal	0.0186 cm	0.0008 cm
Clad thickness	Normal	0.0673 cm	0.0008333333 cm
U235 enrichment	Normal	4.85 %	0.07466 %

Exercise I-1

Exercise I-1, Cell Physics, is focused on the derivation of the multi-group microscopic cross-section libraries

- **Exercise I-1 propagates the uncertainties in evaluated Nuclear Data Libraries - NDL - (microscopic point-wise cross sections) into multi-group microscopic cross-sections used as an input by lattice physics codes and associated multi-group covariance matrices**
- **The participants can use any of the major NDLs such as ENDF, JEFF, and JENDL**
- **The development of nuclear data covariance files is in progress in major NDLs**

For the purposes of the OECD LWR UAM benchmark the availability of relative covariance data is important for:

- **All relevant nuclides (actinides, fission products, absorbers and burnable poisons, structural materials and etc.) present in the reactor core and reflector regions of LWRs**
- **Covering the entire energy range of interest (from 0 to 10 MeV)**
- **All relevant reaction cross-section types**

Exercise I-1

Number of materials and cross-sections with covariances of neutron cross-sections

Data files	*Number of materials*	*Number of cross-sections*
ENDF/B-VI.8	44	400
JEFF-3.1	34	350
JENDL-3.3	20	160
TENDL-2008	from F-19-Po-209	all

Number of nuclides and energy groups in the available multi-group covariance matrices

Name	*Number of nuclides*	*Number of energy groups*
ANL	42	17
NEA/OECD	31	15
SCALE5.1/ORNL SCALE6.0/ORNL	299 401	44 44

Exercise I-1

- The current status of the evaluated cross-section NDLs is such that the most comprehensive covariance library is available with SCALE-5.1 and now with the extension / improvement in SCALE-6.0
- For this reason initially it was decided to utilize the nuclide dependent multi-group covariance data from SCALE-6.0 for the purposes of Exercise I-1
- It is based on a 44-group structure. For other group structures, NEA/OECD has provided the tools for handling and transforming the cross-section covariance in a consistent way (ANGELO and LAMBDA)
- Covariance data are relative values and can be used with different NDLs
- SCALE 6.0 covariance library + updated version of ANGELO and LAMBDA are delivered to all participants
- SCALE 6.1 with GPT implemented is now available for the participants
- Collaboration of the UAM benchmark group with ORNL:
    - Early access to Beta version of SCALE-6.2;
    - Access to standardized interface of SCALE to different sampling tools;
    - Interfaces to core simulators.

Exercise I-1

- The data in the SCALE-6 library has been assembled from a variety sources
- Includes recent ENDF/B-VII covariance evaluations
- Approximate uncertainties span full energy range 0-20 MeV
- Approximate uncertainties included for all the reaction cross-sections for all materials present in LWR cores
- Includes uncertainties in the fission spectra which is very important in multi-group reactor calculations;
- The 44-group covariance library is “generic” (problem-independent) and thus the participants have to address problem-specific resonance self-shielding effects for performing sensitivity and uncertainty analysis
- Only energy is fully correlated in the SCALE-6.0 44-group covariance library. There are no real cross-reaction and cross-nuclide correlations

Exercise I-1

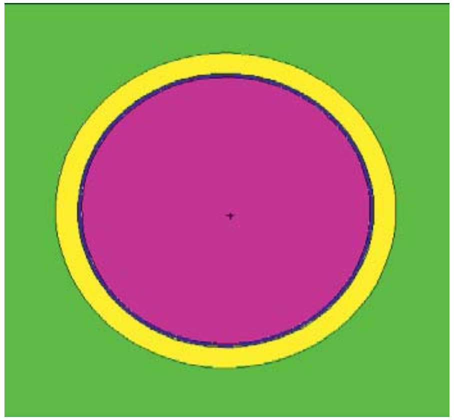
H-1, H-ZrH, H-poly, H-freegas, H-2, H2-freegas, H-3, He-3, He-4, Li-6, Li-7, Be-7, Be-9, Be-bound, B-10, B-11, C-0, C-graphite, N-14, N-15, O-16, O-17, F-19, Na-23, Mg-0, Mg-24, Mg-25, Mg-26, Al-27, Si-0, Si-28, Si-29, Si-30, P-31, S-0, S-32, S-34, S-36, Cl-0, Cl-35, Cl-37, Ar-36, Ar-38, Ar-40, K-0, K-39, K-40, K-41, Ca-0, Ca-40, Ca-42, Ca-43, Ca-44, Ca-46, Ca-48, Sc-45, Ti-0, Ti-46, Ti-47, Ti-48, Ti-49, Ti-50, V-0, Cr-50, Cr-52, Cr-53, Cr-54, Mn-55, Fe-0, Fe-54, Fe-56, Fe-57, Fe-58, Co-58, Co-58(m), Co-59, Ni-58, Ni-59, Ni-60, Ni-61, Ni-62, Ni-64, Cu-63, Cu-65, Ga-0, Ga-69, Ga-71, Ge-70, Ge-72, Ge-73, Ge-74, Ge-76, As-74, As-75, Se-74, Se-76, Se-77, Se-78, Se-79. Se-80, Se-82, Br-79, Br-81, Kr-78, Kr-80, Kr-82, Kr-83, Kr-84, Kr-85, Kr-86, Rb-85, Rb-86, Rb-87, Sr-84, Sr-86, Sr-87, Sr-88, Sr-89, Sr-90, Y-89, Y-89, Y-90, Y-91, Zr-0, Zr-90, Zr-91, Zr-92, Zr-93, Zr-94, Zr-95, Zr-96, Nb-93, Nb-94, Nb-95, Mo-0, Mo-92, Mo-94, Mo-95, Mo-96, Mo-97, Mo-98, Mo-99, Mo-100, Tc-99, Ru-96, Ru-98, Ru-99, Ru-100, Ru-101, Ru-102, Ru-103, Ru-104, Ru-105, Ru-106, Rh-103, Rh-105, Pd-102, Pd-104, Pd-105, Pd-106, Pd-107, Pd-108, Pd-110, Ag-107, Ag-109, Ag-111, Cd-0, Cd-106, Cd-108, Cd-110, Cd-111, Cd-112, Cd-113, Cd-114, Cd-115(m), Cd-116, In-0, In-113, In-115, Sn-112, Sn-113, Sn-114, Sn-115, Sn-116, Sn-117, Sn-118, Sn-119, Sn-120, Sn-122, Sn-123, Sn-124, Sn-125, Sb-121, Sb-123, Sb-124, Sb-125, Sb-126, Te-120, Te-122, Te-123, Te-124, Te-125, Te-126, Te-127(m), Te-128, Te-129(m), Te-130, I-127, I-129, I-130, I-131, I-135, Xe-123, Xe-124, Xe-126, Xe-128, Xe-129, Xe-130, Xe-131, Xe-132, Xe-133, Xe-134, Xe-135, Xe-136, Cs-133, Cs-134, Cs-135, Cs-136, Cs-137, Ba-130, Ba-132, Ba-133, Ba-135, Ba-136, Ba-137, Ba-138, Ba-140, La-138, La-139, La-140, Ce-136, Ce-138, Ce-139, Ce-140, Ce-141, Ce-142, Ce-143, Ce-144, Pr-141, Pr-142, Pr-143, Nd-142, Nd-143, Nd-144, Nd-145, Nd-146, Nd-147, Nd-148, Nd-150, Pm-147, Pm-148, Pm-148(m), Pm-149, Pm-151, Sm-144, Sm-147, Sm-148, Sm-149, Sm-150, Sm-151, Sm-152, Sm-153, Sm-154, Eu-151, Eu-152, Eu-153, Eu-154, Eu-155, Eu-156, Eu-157, Gd-152, Gd-153, Gd-154, Gd-155, Gd-156, Gd-157, Gd-158, Gd-160, Tb-159, Tb-160, Dy-156, Dy-158, Dy-160, Dy-161, Dy-162, Dy-163, Dy-164, Ho-165, Er-162, Er-164, Er-166, Er-167, Er-168, Er-170, Lu-175, Lu-176, Hf-0, Hf-174, Hf-176, Hf-177, Hf-178, Hf-179, Hf-180, Ta-181, Ta-182, W-0, W-182, W-183, W-184, W-186, Re-185, Re-187, Ir-191, Ir-193, Au-197, Hg-196, Hg-198, Hg-199, Hg-200, Hg-201, Hg-202, Hg-204, Pb-204, Pb-206, Pb-207, Pb-208, Bi-209, Ac-225, Ac-226, Ac-227, Th-227, Th-228, Th-229, Th-230, Th-232, Th-233, Th-234, Pa-231, Pa-232, Pa-233, U-232, U-233, U-234, U-235, U-235, U-236, U-237, U-238, U-239, U-240, U-241, Np-235, Np-236, Np-237, Np-238, Pu-236, Pu-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Pu-243, Pu-244, Pu-246, Am-241, Am-242, Am-242(m), Am-243, Am-244, Cm-241, Cm-242, Cm-243, Cm-244, Cm-245, Cm-246, Cm-247, Cm-248, Cm-249, Cm-250, Bk-249, Bk-250, Cf-249, Cf-250, Cf-251, Cf-252, Cf-253, Cf-254, Es-253, Es-254, Es-255, Fm-255

In red : added nuclides / materials

21

Exercise I-1

- In order to perform a comparative analysis of the multi-group cross-section uncertainty data obtained after processing test problems are devised or utilized from the previously defined benchmarks and from the available experimental data
- The first sets of problems are two-dimensional fuel pin-cell test problems representative of BWR PB-2, PWR TMI-1, and VVER-1000 Kozloduy-6
- These problems are analyzed at Hot Zero Power (HZP) conditions and Hot Full Power (HFP) conditions
- The BWR HFP case is at 40 % void is added for different spectrum conditions
- Two critical experiments:
  - KRITZ 2.1 (UOX)
  - KRITZ 2.13 (UOX)
  - KRITZ 2.19 (MOX)



MCNP5 2-D model of the PB-2 BWR pin cell

Three types of fuel composition from a representative Generation III LWR specification (provided by CEA, France) are analyzed at Hot Full Power condition:

- ✓ UOX
- ✓ MOX
- ✓ UOX-Gd_2O_3

For each type of unit cell uncertainty in k_{inf} for different enrichments of fissile material are compared

Exercise I-1

Test cases which cover extensive range of materials and temperature (void) and spectrum conditions:

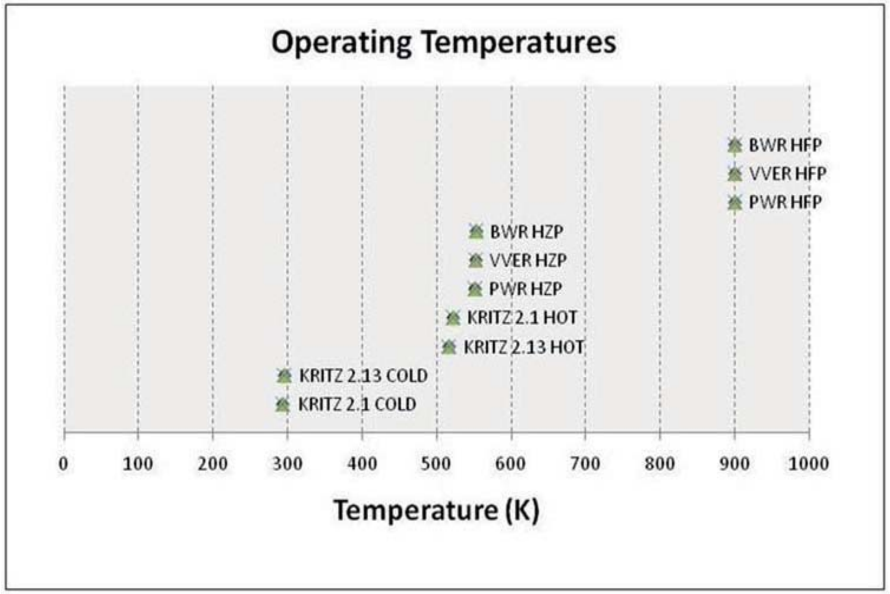
- ✓ Analysis of temperature effect on uncertainty in k_{inf} for selected unit fuel pin cells
- ✓ Analysis of composition effect on uncertainty in k_{inf} for selected unit fuel pin cells

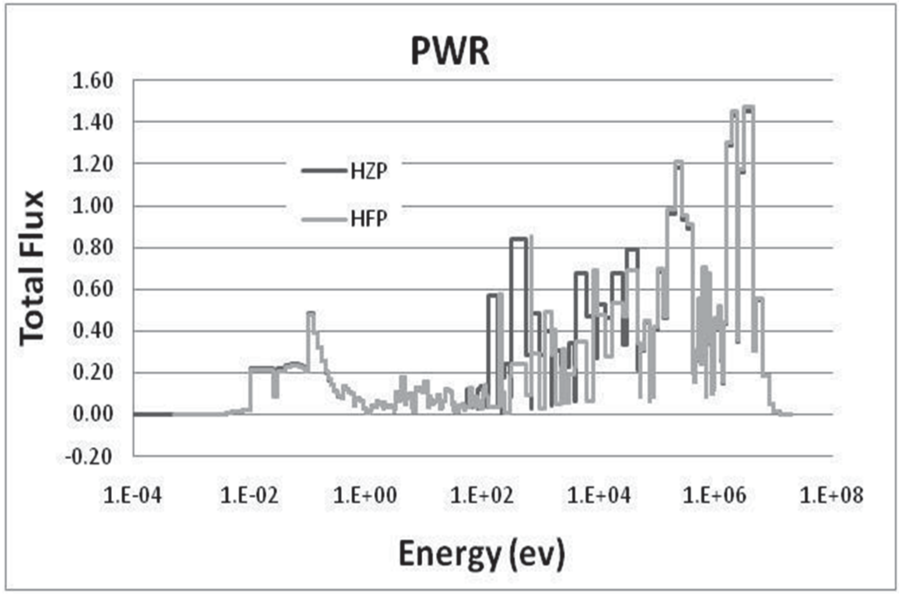
Fuel	Operating Conditions	K_{eff}	Uncertainty in K_{eff} (% Δ-k/k)	Largest Nuclide Reaction Cross-section Contributor to Uncertainty
BWR	HZP	1.339032	0.5289	^{238}U (n,γ)
	HFP	1.324305	0.5293	^{238}U (n,γ)
PWR	HZP	1.421857	0.4817	^{238}U (n,γ)
	HFP	1.403147	0.4957	^{238}U (n,γ)
VVER	HZP	1.364097	0.5145	^{238}U (n,γ)
	HFP	1.346205	0.5225	^{238}U (n,γ)
KRITZ-2.1	Cold	1.232409	0.5874	^{238}U (n,γ)
	Hot	1.184100	0.6309	^{238}U (n,γ)
KRITZ-2.13	Cold	1.265097	0.5458	^{238}U (n,γ)
	Hot	1.234826	0.5757	^{238}U (n,γ)

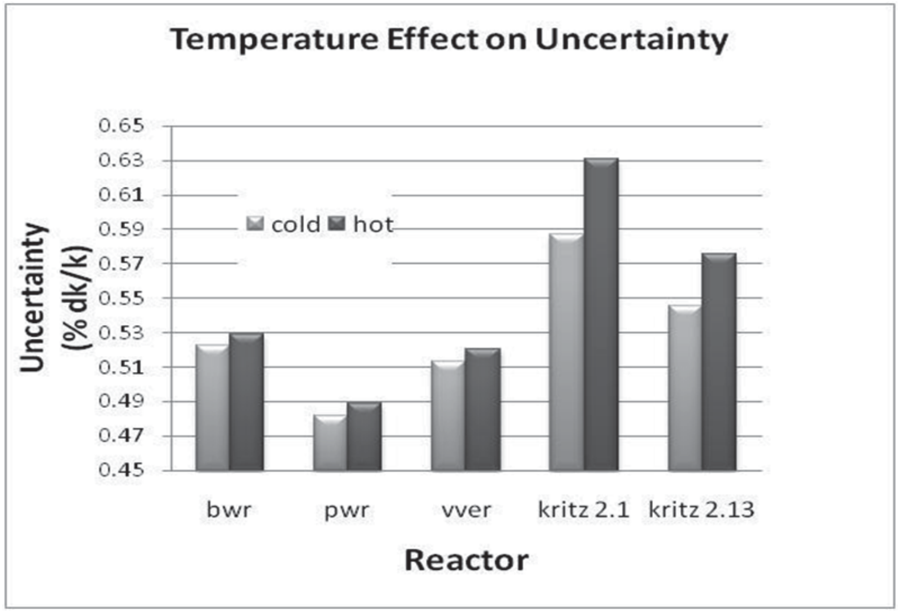
Fuel	Compositions	K_{eff}	Uncertainty in K_{eff} (% Δ-k/k)	Largest Nuclide Reaction Cross-section Contributor to Uncertainty
MOx	9.8% ^{239}Pu	1.095097	0.9398	^{238}U (n,n')
	6.5% ^{239}Pu	1.057435	0.9651	^{238}U (n,n')
	3.7% ^{239}Pu	1.014537	0.9877	^{238}U (n,n')
UOx	4.2% ^{235}U	1.245378	0.5115	^{238}U (n,γ)
	3.2% ^{235}U	1.176433	0.5367	^{238}U (n,γ)
	2.1% ^{235}U	1.051125	0.5910	^{238}U (n,γ)
UOxGd$_2$O$_3$	2.2% ^{235}U	0.216013	1.7667	^{238}U (n,n')
	1.9% ^{235}U	0.199026	1.9334	^{238}U (n,n')

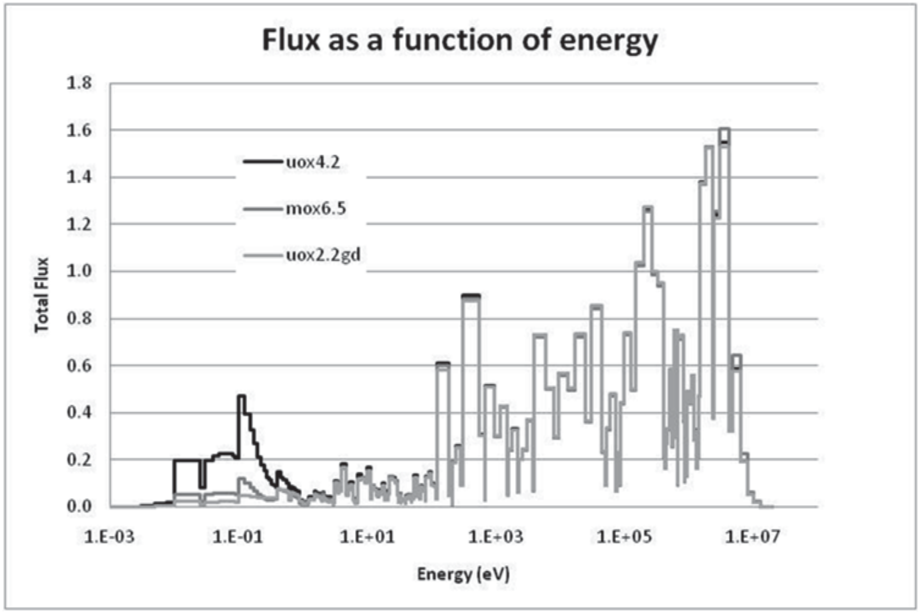
Source UPC and PSU

Exercise I-1


Operating Temperatures
BWR HFP
VVER HFP
PWR HFP
BWR HZP
VVER HZP
PWR HZP
KRITZ 2.1 HOT
KRITZ 2.13 HOT
KRITZ 2.13 COLD
KRITZ 2.1 COLD
0
100
200
300
400
500
600
700
800
900
1000
Temperature (K)


PWR
1.60
1.40
1.20
1.00
0.80
0.60
0.40
0.20
0.00
-0.20
HZP
HFP
Total Flux
1.E-04
1.E-02
1.E+00
1.E+02
1.E+04
1.E+06
1.E+08
Energy (ev)


Temperature Effect on Uncertainty
0.65
0.63
0.61
0.59
0.57
0.55
0.53
0.51
0.49
0.47
0.45
cold
hot
Uncertainty (% dk/k)
bwr
pwr
vver
kritz 2.1
kritz 2.13
Reactor


Flux as a function of energy
1.8
1.6
1.4
1.2
1.0
0.8
0.6
0.4
0.2
0.0
uox4.2
mox6.5
uox2.2gd
Total Flux
1.E-03
1.E-01
1.E+01
1.E+03
1.E+05
1.E+07
Energy (eV)

Exercise I-1

- Fuel pin-cell test problems from the KRITZ-2 LEU and MOX critical experiments
- The KRITZ-2:1 and KRITZ-2:13 experiments at two different temperatures and boron concentration are selected since their rod pitch sizes are similar to those of lattices present in the PB-2 and TMI-1 cores
- The KRITZ-2:19 experiment is a representative of a MOX lattice and also is analyzed
- For each test problem and case participants have to calculate k_{inf}, and absorption and fission reaction rates for ^{234}U, ^{235}U, and ^{238}U and associated uncertainties

Based on
SCALE-6

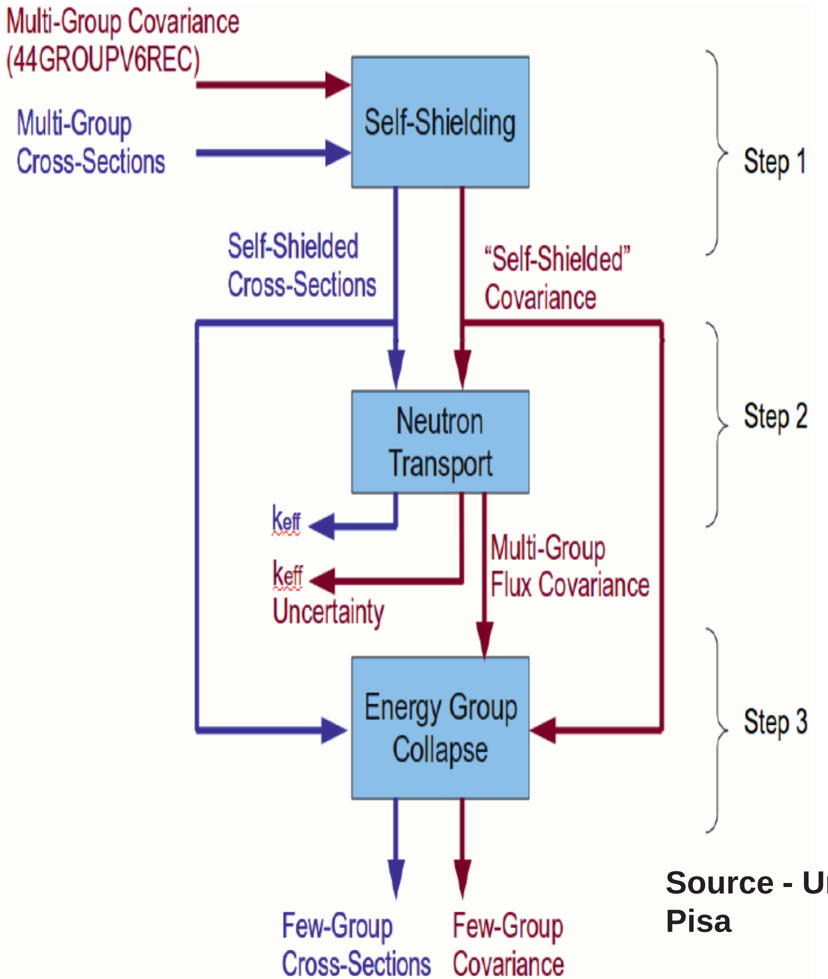
Source - IJS

Exercise I-2

Exercise I-2, Lattice Physics, is focused on the derivation of the few-group macroscopic cross-section libraries

- **In the current established calculation scheme for LWR design and safety analysis, multi-group microscopic cross-section libraries are an input to lattice physics calculations**

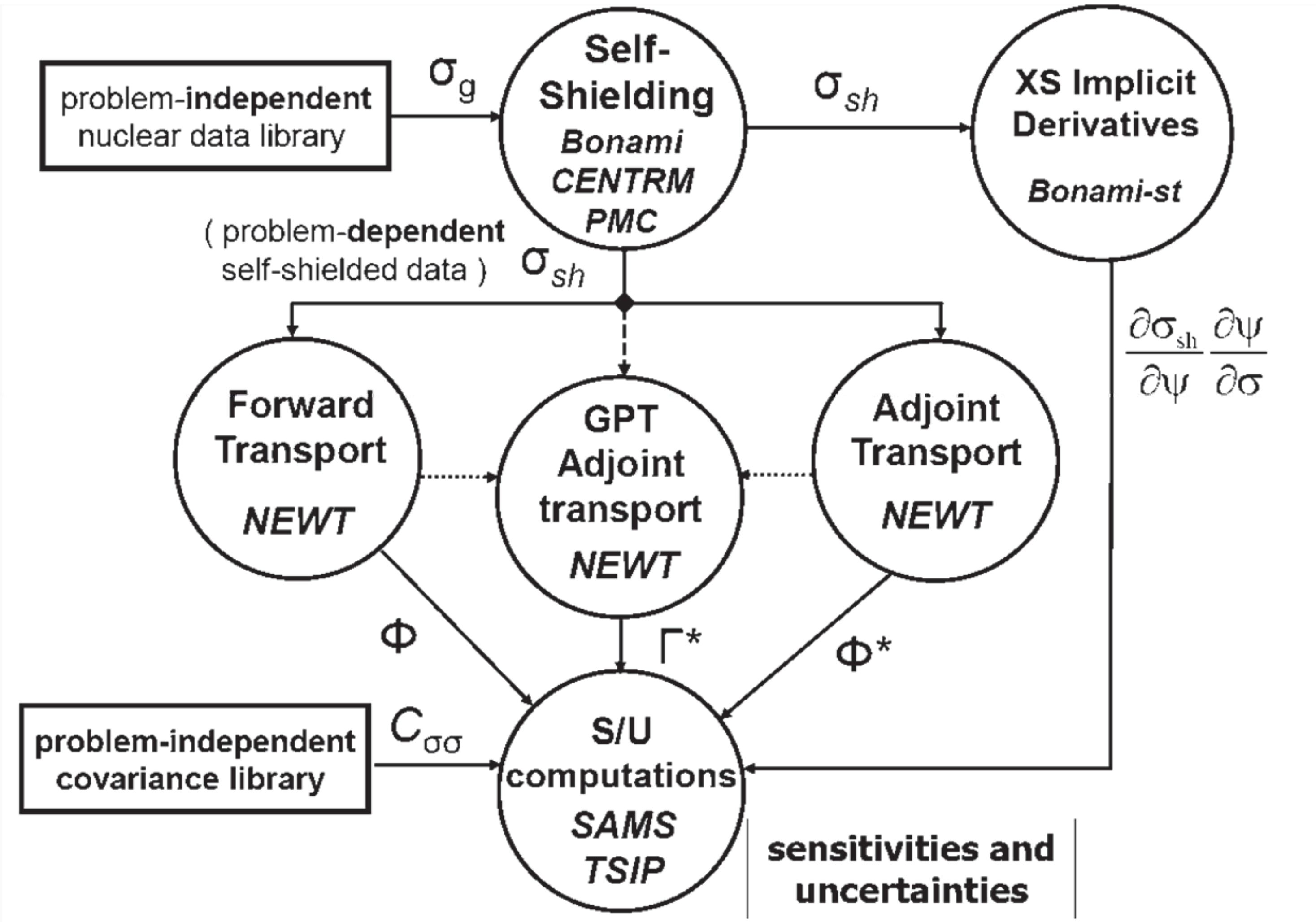
- **The multi-group cross-section uncertainties (multi-group cross-section covariance matrix) should be obtained by participants as output uncertainties within the framework of Exercise I-1**



Source - University of Pisa

Exercise I-2

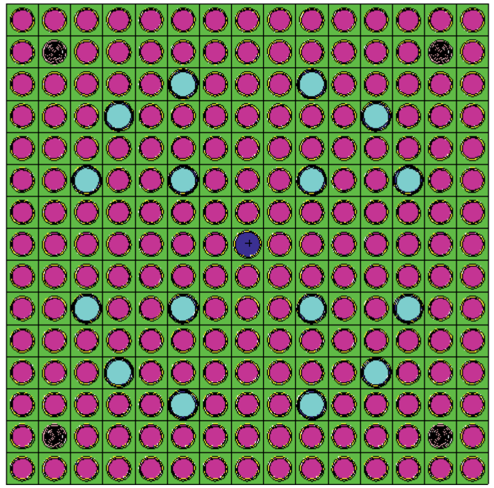
GPT Sequence in TSUNAMI (available with SCALE-6.1)



Source - ORNL

Exercise I-2

- The first set of problems are test problems representative of BWR PB-2, PWR TMI-1, and VVER-1000 Kozloduy-6 defined on assembly spatial scale
- These problems are analyzed at Hot Zero Power (HZP) conditions and Hot Full Power (HFP) conditions to account for spectrum changes. For BWR case also different void fraction conditions are considered
- Continuous energy Monte Carlo reference solutions The second set of problems are based on publically available experimental data



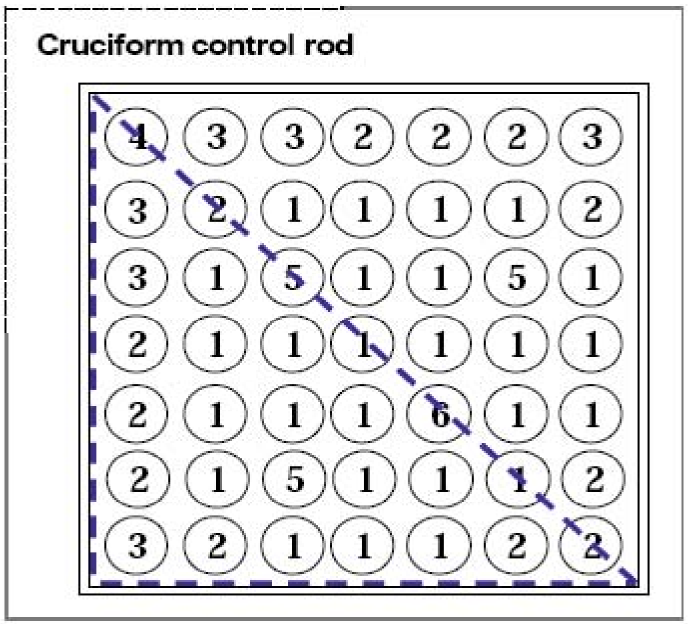
TMI MCNP5 Assembly Model

Assembly	Nuclear Data Library	Thermal Scattering Library	PURR k_{inf}/Std.		GROUPR k_{inf}/Std.	
TMI	ENDF/B-VII.0	lwtr.62t	1.05899	0.00016	1.05826	0.00015
		lwtr.04t	1.05737	0.00016	1.05718	0.00016
		th552.68t	1.06239	0.00025	1.06212	0.00025
	JEFF3.1.1	lwtr.62t	1.05490	0.00015	1.05409	0.00015
		lwtr.04t	1.05359	0.00015	1.05253	0.00016
		th552.68t	1.05851	0.00023	1.05746	0.00025
	JENDL3.1	lwtr.62t	1.05544	0.00025	1.05511	0.00023
		th552.68t	1.05924	0.00026	1.05849	0.00024

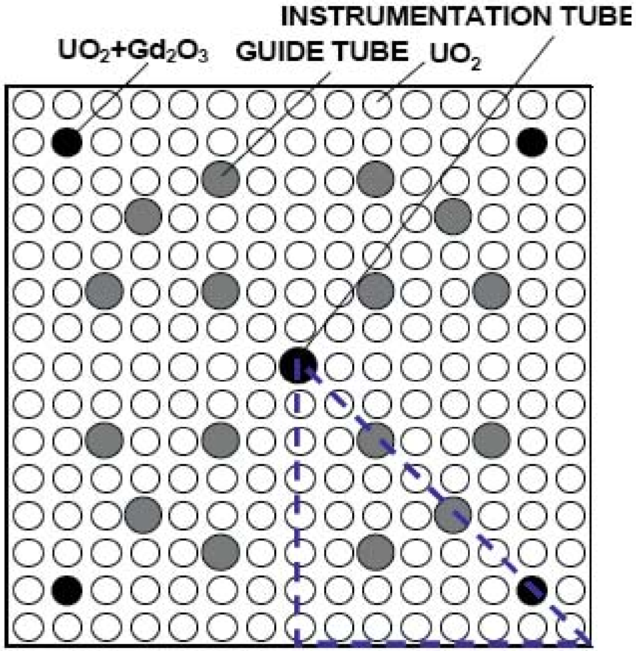
Exercise I-2

Fission rate and uncertainties

PB-2 assembly



TMI-1 assembly



Source - JNES

Exercise I-2

Fission rate for the fuel pins and uncertainties for PB-2

unrodded case

Fission rate [-]

1.122						
1.168	1.025					
1.047	1.177	0.275				
1.117	1.129	0.877	0.833			
1.113	1.126	0.873	0.803	0.265		
1.162	1.17	0.277	0.884	0.948	1.121	
1.158	1.027	1.183	1.158	1.226	0.986	1.143

Uncertainties [%]

0.552						
0.551	0.544					
0.547	0.546	0.451				
0.548	0.544	0.534	0.532			
0.548	0.544	0.534	0.531	0.449		
0.550	0.546	0.454	0.535	0.538	0.544	
0.551	0.545	0.548	0.546	0.548	0.543	0.549

□ : $UO_2+Gd_2O_3$

Exercise I-2

Fission rate of the fuel pins and uncertainties for TMI-1

unrodded case

Fission rate [-]

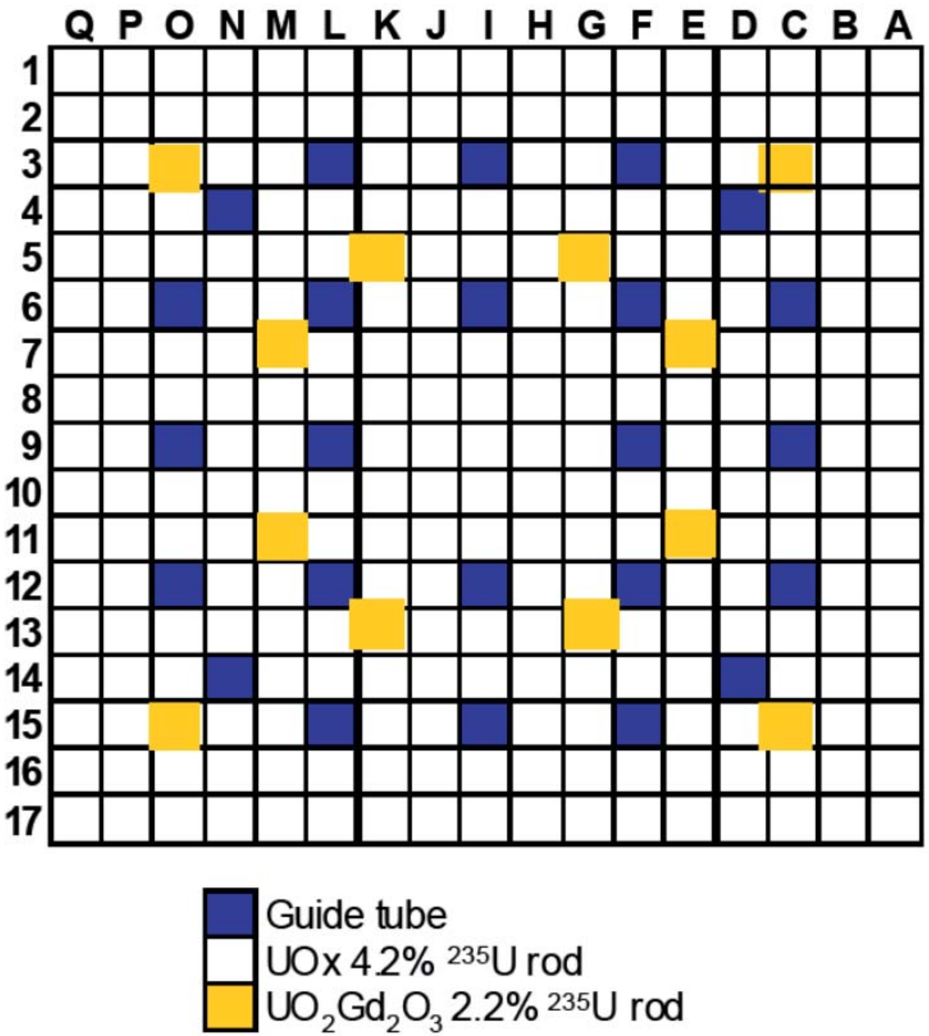
1.085	1.066						
1.027	1.081						
1.000	1.033	1.096	1.082				
0.998	1.026	1.096	1.115				
0.998	1.058		1.093	1.049	0.953		
0.982	1.008	1.057	1.006	0.957	0.879	0.332	
0.996	0.999	1.004	0.987	0.958	0.913	0.868	0.915

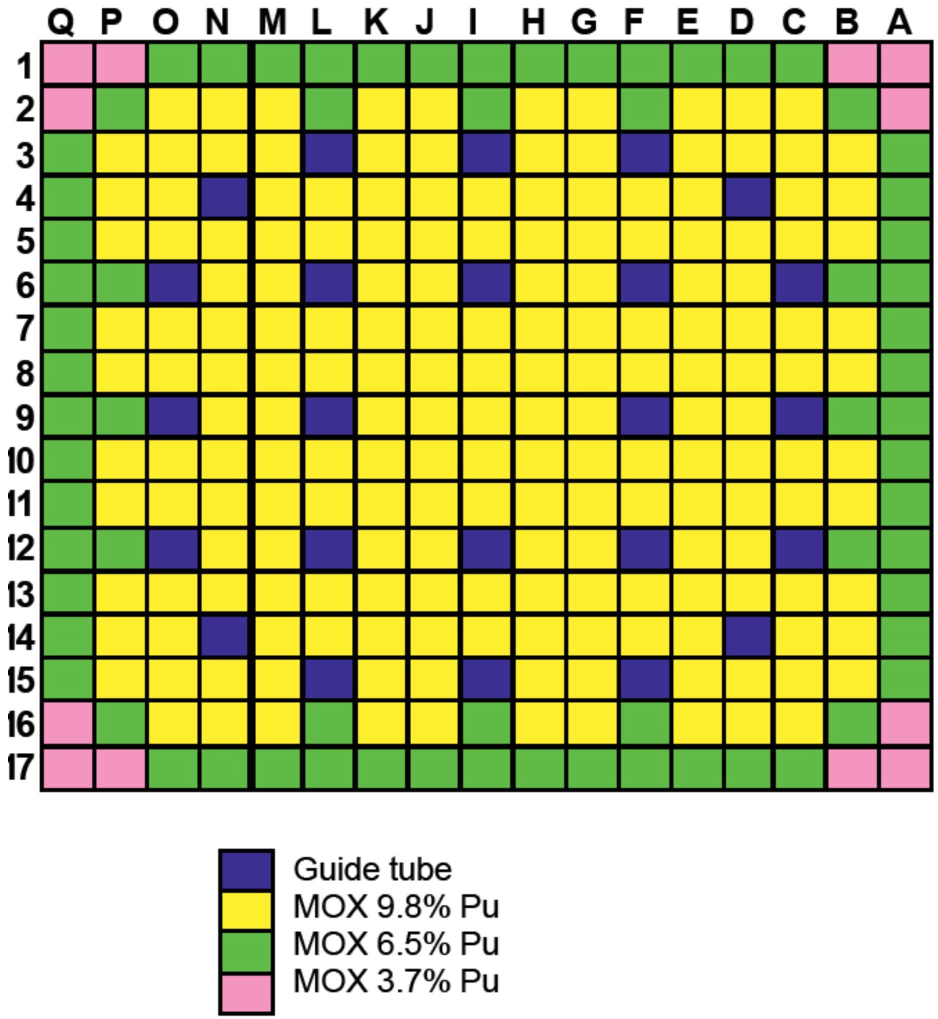
Uncertainties [%]

0.546	0.545						
0.543	0.545						
0.542	0.543	0.546	0.546				
0.542	0.543	0.546	0.547				
0.542	0.545		0.546	0.545	0.541		
0.541	0.543	0.545	0.543	0.541	0.538	0.480	
0.542	0.542	0.543	0.542	0.542	0.540	0.538	0.541

□:UO_2+Gd_2O_3, ▨:Guide,Instrumentation tube

Exercise I-2





Specification of PWR Generation III assemblies
Source: CEA

Exercise I-3

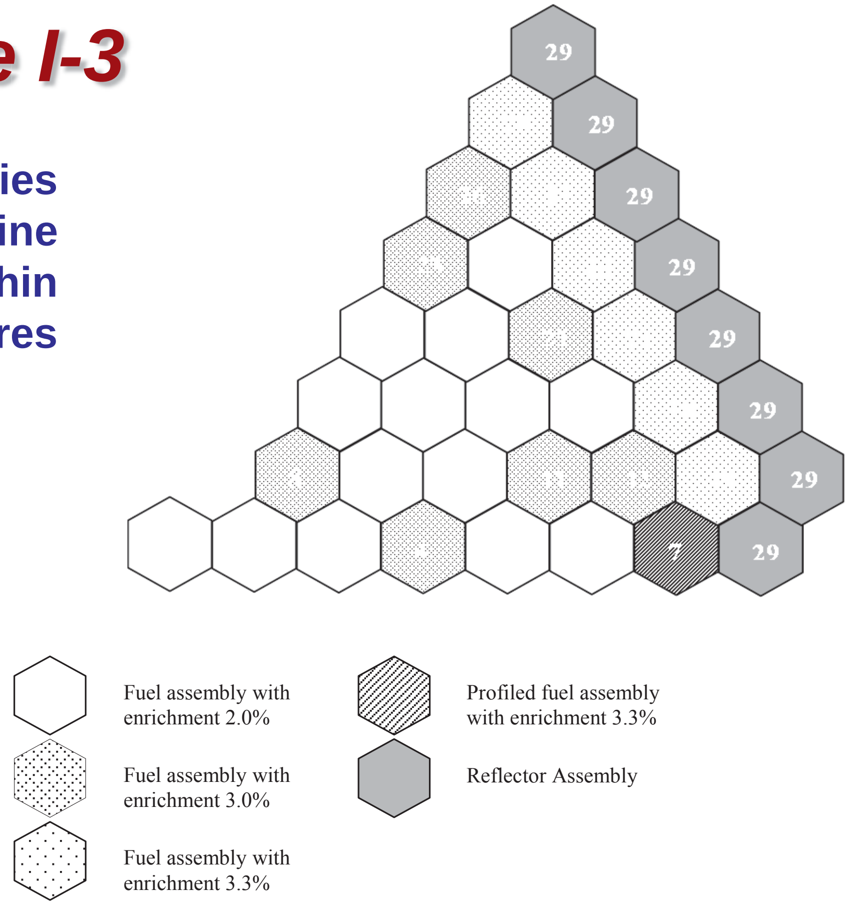
Exercise I-3, Core Physics, is focused on the core steady state stand-alone neutronics calculations

- **In the current established calculation scheme for LWR design and safety analysis the lattice averaged (homogenized) few-group cross-sections are an input to core calculations**
- **The few-group cross-section uncertainties (few-group covariance matrix) should be obtained by participants as output uncertainties within the framework of Exercise I-2**
- **In Exercise I-3 the few-group cross-section uncertainties are input uncertainties and must be propagated to uncertainties in evaluated stand-alone neutronics core parameters**

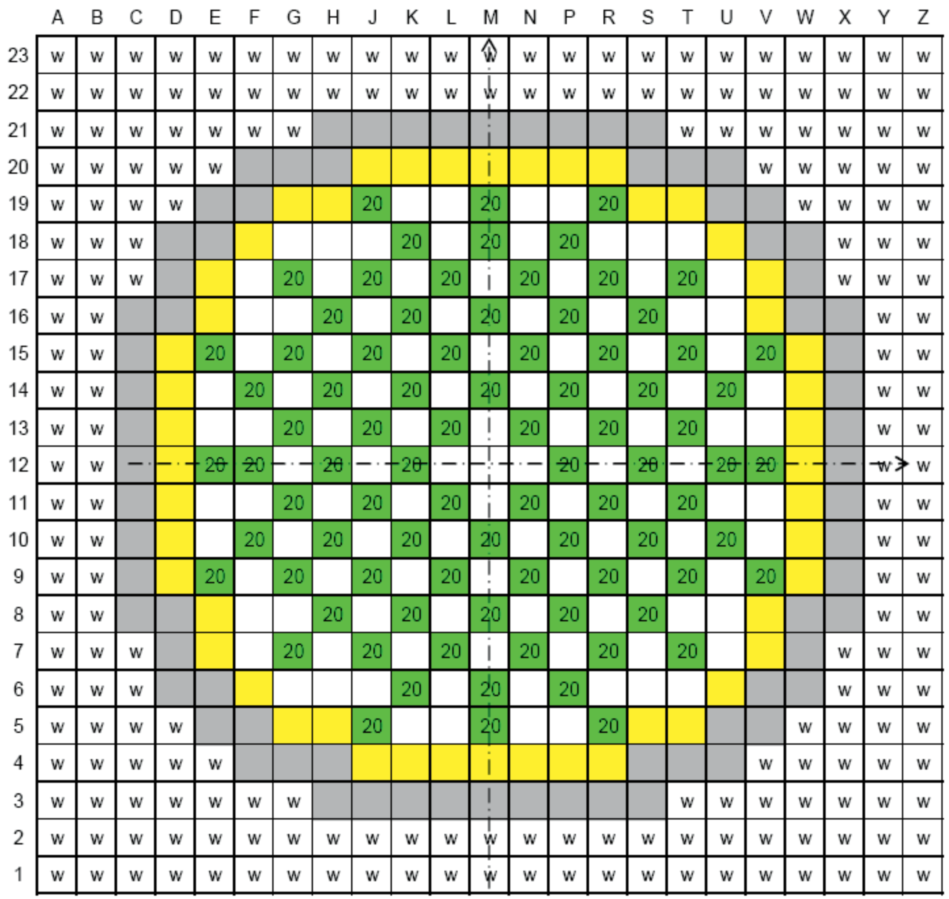
PB-2 BWR HZP case

Exercise I-3

- The propagation of the input uncertainties through core calculations to determine uncertainties in output core parameters within the framework of Exercise I-3 requires utilization of a core simulator code



VVER-1000 Core

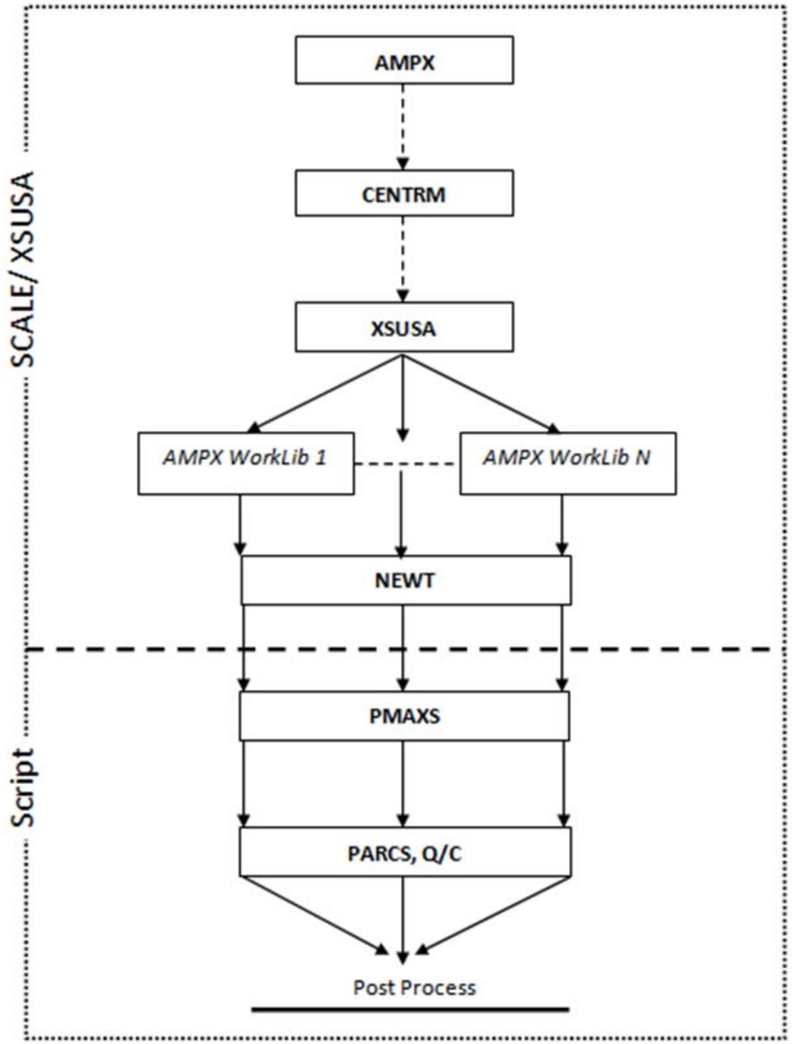


Generation III PWR MOX core
Source - CEA

Phase I

- Test problems on two different levels are defined to be used within Phase I of the OECD LWR UAM benchmark:
  - HZP test cases based on the realistic LWR designs (for which the continuous energy Monte Carlo method is used for reference calculations)
  - Documented experimental benchmark plant cold critical data and critical lattice data
- In summary, Phase I is focused on stand-alone neutronics core calculations and associated prediction uncertainties
- It does not analyze uncertainties related to cycle and depletion calculations
- No feedback modelling is assumed:
  - It will address the propagation of uncertainties associated with few-group cross-section generation, but will not address cross-section modelling (it will be addressed in the following Phases)

XSUSA Method with SCALE/ PARCS

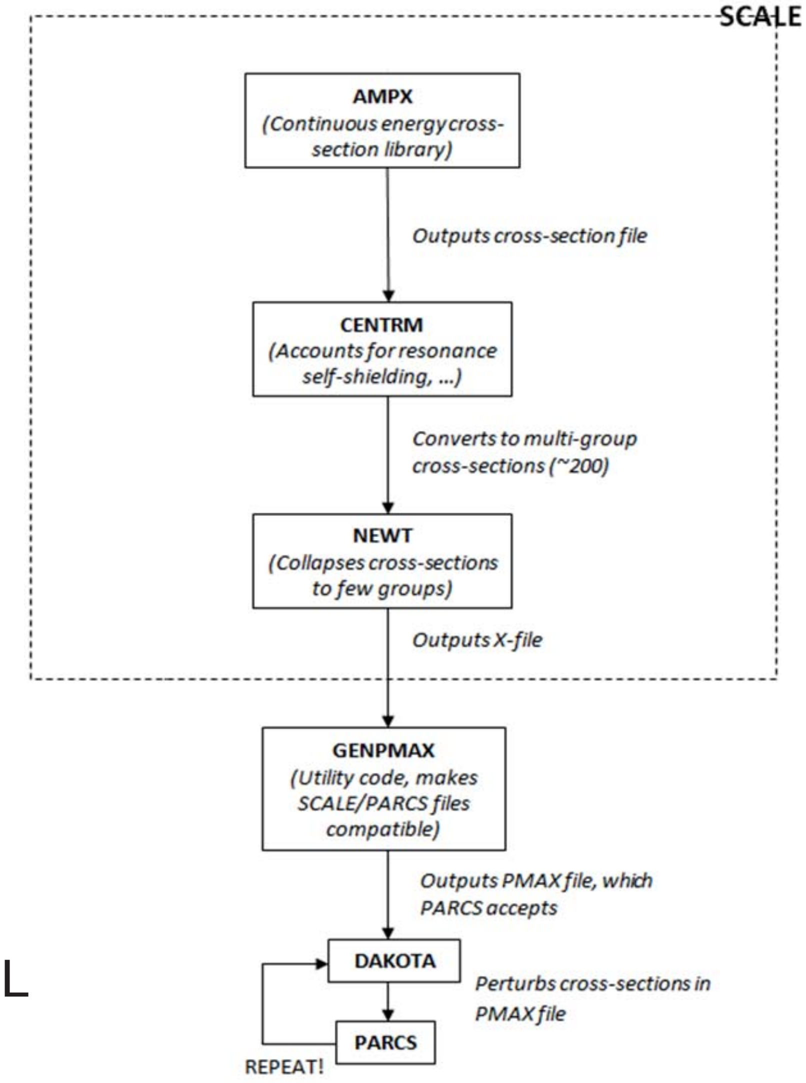


Source – ORNL, UM and GRS

Phase I

- **Primary differences between the random sampling and two-step methods are the moment at which the perturbations are applied and the origin of the covariance matrix**
- **Few-group covariance matrix is obtained using GPT through NEWT.**
- **Few-group covariance matrix is then sampled using the uncertainty software package DAKOTA**

Two-Step Method Using GPT



Source - ORNL

Phase I

Infinite TMI-1 PWR mini-core

U	U	U
U	R	U
U	U	U

TMI Minicore with no Reflector		
Code	**k-eff**	**σ**
PARCS	1.38684	0.00658
Q/C	1.38691	0.00658
KENO	1.38568	0.00664

Source – UM and GRS

Phase I

Reflected TMI-1 PWR mini-core

Source UM and GRS

Reflector	Reflector	Reflector	Reflector	Reflector
Reflector	U	U	U	Reflector
Reflector	U	R	U	Reflector
Reflector	U	U	U	Reflector
Reflector	Reflector	Reflector	Reflector	Reflector

TMI Minicore with Reflector		
Code	**k-eff**	**σ**
PARCS	1.15244	0.00578
Q/C	1.15434	0.00583
Difference (pcm)	190	5

Uncertainties of Kinetics Parameters

- ✓ SNEAK (fast core problem) was added as an optional test case to the test problems for Exercise I-3 since it has a unique set of experimental data for β_{eff} uncertainties and can be used as an example on how to calculate uncertainty in β_{eff}.
- ✓ The two high-quality reactor physics benchmark experiments, SNEAK-7A & 7B (Karlsruhe Fast Critical Facility) are part of the International Reactor Physics Benchmark Experiments (IRPhE) database.
- ✓ It was demonstrated that the energy field responsible for β_{eff} uncertainty is the same for fast and thermal reactors => SNEAK cases are relevant to the any kinds of kinetic parameters calculations validation

Parameters	SNEAK 7A	SNEAK 7B
Λ, μs	0.180	0.159
Λ, uncertainty	4.15%	3.21%
β	0.00395	0.00429
β, direct uncertainty	2.0 %	2.5%
β, uncertainty	2.4%	N/A

Source - IRSN

Participation in Phase I

Organization	Country	Exercise I-1	Exercise I-2	Exercise I-3
PSU/UPC	USA/Spain	X	X	X
MkMaster/UPisa	Canada/Italy	X	X	
ORNL	USA	X	X	X
GRS	Germany	X	X	X
PSI	Switzerland	X	X	
VTT	Finland	X	X	
UPM	Spain	X	X	
KFKI	Hungary	X	X	
CEA	France	X	X	X
UM	USA	X	X	X
TUM	Germany	X		
NECSA	South Africa	X	X	
JNES	Japan	X	X	
LPSC/CNRS-PM	Canada/France	X		

Phases II and III

- **The obtained output uncertainties from Phase I of the OECD LWR UAM benchmark will be utilized as input uncertainties in the remaining two phases – Phase II (Core Phase) and Phase III (System Phase)**

- **Phase II will address core neutron kinetics, thermal-hydraulics and fuel performance, without any coupling between the three physics phenomena**

- **Phase III will include system thermal-hydraulics and coupling between fuel, neutronics and thermal-hydraulics for steady-state, depletion and transient analysis**

Phase II

Phase II - Core Phase:

- ✓ Exercise II-1 – Fuel Physics: Fuel thermal properties relevant to steady-state and transient performance

  U-4 (uncertainties in fuel temperature – Doppler feedback)

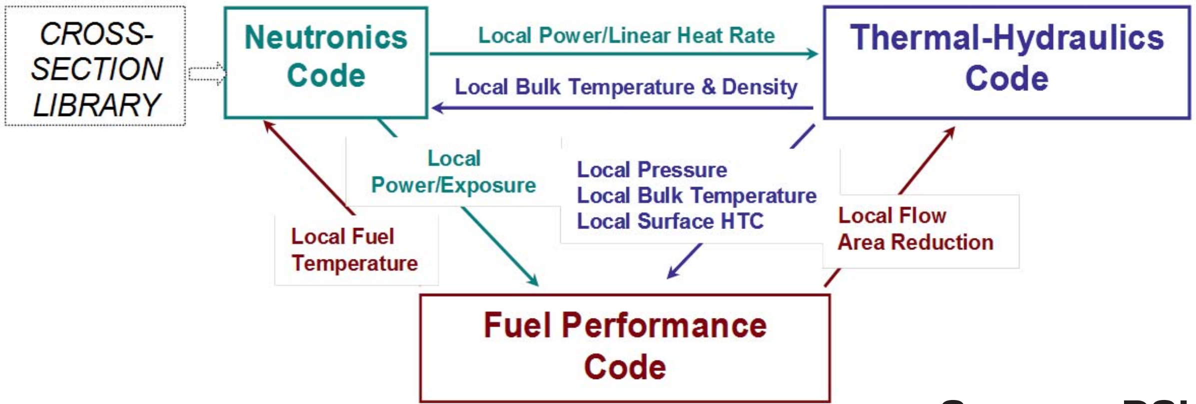
- ✓ Exercise II-2 – Neutron Kinetics : Neutron kinetics stand-alone performance

  U-5 (uncertainties in time-dependent (dynamic) reactivity insertion, total power evolution and power peaking factors)

- ✓ Exercise II-3 – Bundle Thermal-Hydraulics: Thermal-hydraulic fuel bundle performance

  U-6 (uncertainties in moderator temperature, density and void fraction – moderator feedback)

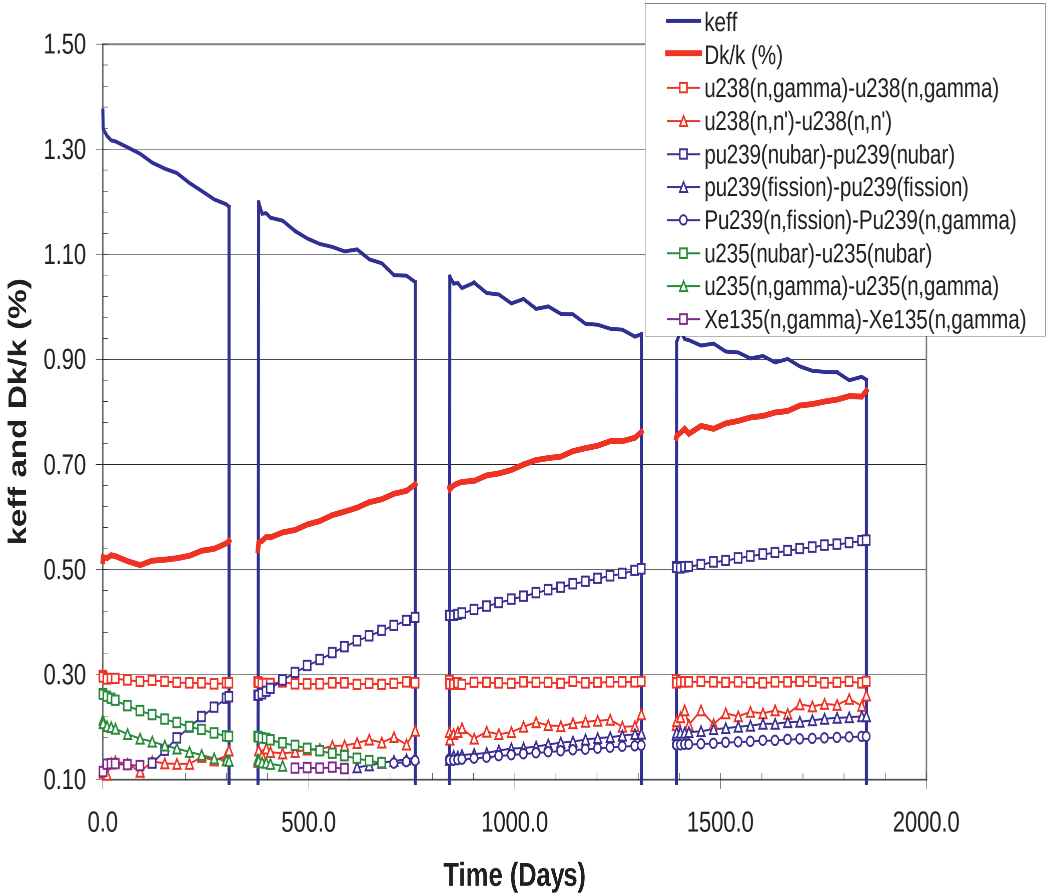
Phase II will address core neutron kinetics, thermal-hydraulics and fuel performance, without any coupling between the three physics phenomena



Source - PSU

Phase II

- Phase II takes into account other physics involved in reactor simulation, i.e. Thermal-Hydraulics and Fuel Physics and introduces time-dependence
- Interaction with Uncertainty Analysis Exercises of the OECD/NRC BFBT and PSBT benchmarks
- Content of Phase II:
  - Exercise II-1 - Fuel Physics
    - ✓ Steady State - Exercise II-1a
    - ✓ Transient - Exercise II-1b
  - Exercise II-2 – Time-dependent Neutronics
    - ✓ Depletion – Exercise II-2a
    - ✓ Neutron Kinetics – Exercise II-2b
  - Exercise II-3 – Bundle Thermal-Hydraulics
    - ✓ Steady State – Exercise II-3a
    - ✓ Transient – Exercise II-3b

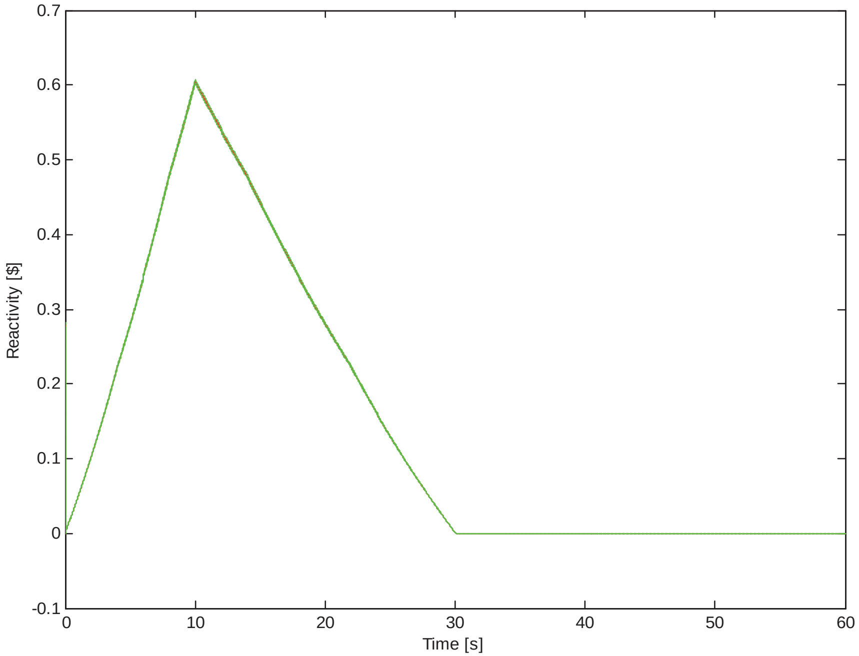


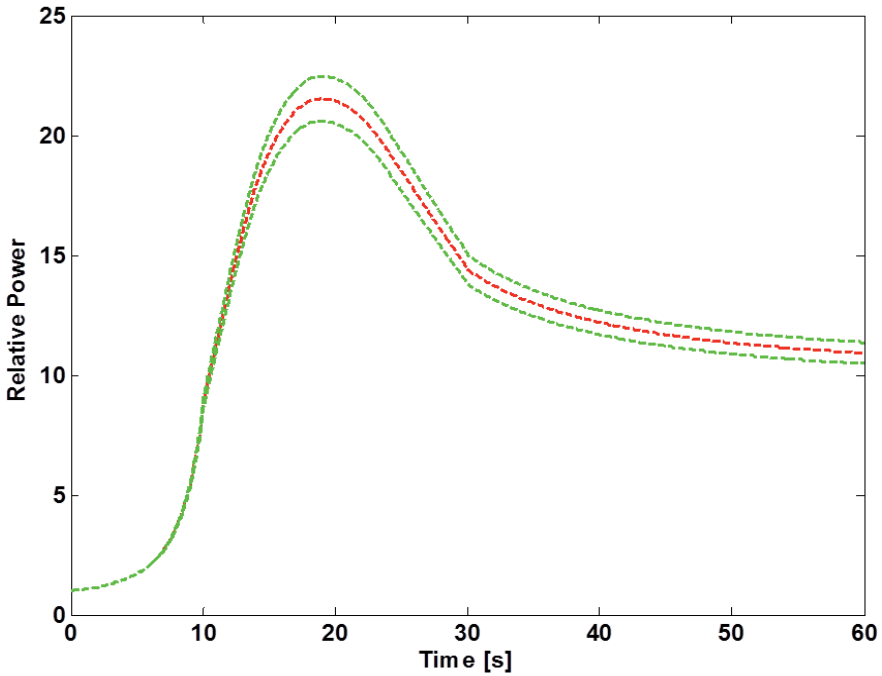
dk/k (%) and the most important contributions with depletion for TMI-1 HFP cell

Source - UPM

Exercise II-2

- A 20 cm tall version of the Reflected TMI-1 PWR Mini-core
- Withdraw control rod 5 cm in 10 seconds ~ $0.6 reactivity insertion
- Reinsert control rod in 20 seconds / 100 simulations





Source - UM

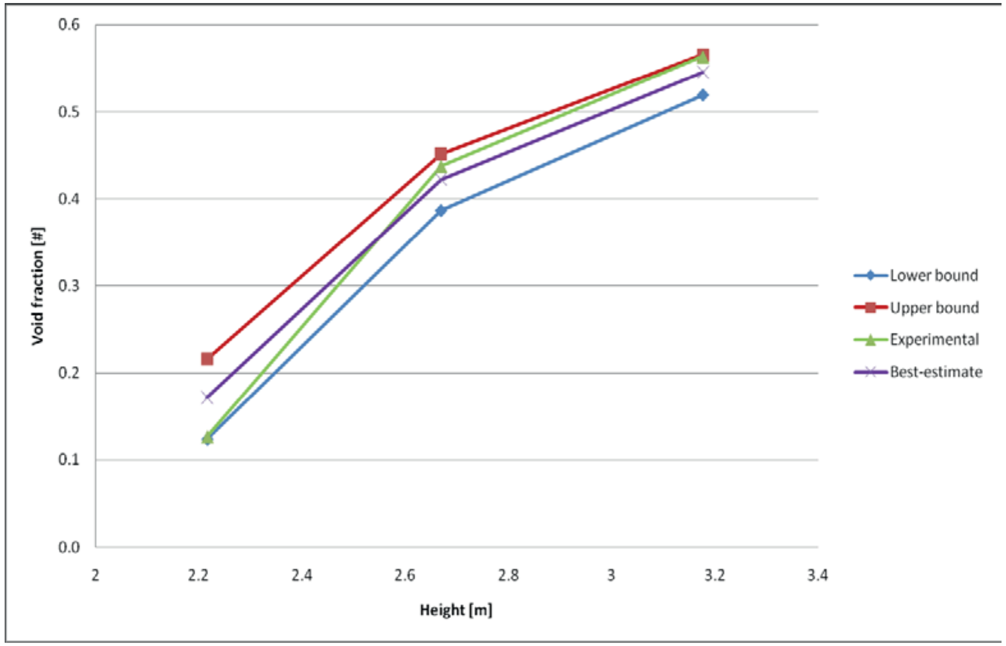
Exercise II-3

Input (I) – boundary conditions, power shapes, geometry, and modeling parameters

Output (O) – pressure drop, CHF/DNB, moderator density, temperature and void distribution

Propagated uncertainty parameters (U) – moderator density, temperature and void distribution

Assumptions (A) – stand-alone T-H steady state and transient modeling



COBRA-3C and DAKOTA application to PWR bundle
Source – TRACTEBEL Engineering

Quantity	Accuracy
Void fraction measurement	
CT measurement	
Gamma-ray beam width	1 mm
Subchannel averaged (steady-state)	3% void
Spatial resolution of one pixel	0.5 mm
Chordal measurement	
Gamma-ray beam width (center)	3 mm
Gamma-ray beam width (side)	2 mm
Subchannel averaged (steady-state)	4% void
Subchannel averaged (transient)	5% void

Estimated Accuracy for Void Fraction Measurements in the PSBT database

Exercise II-3

COBRA-TF and SUSA application to BWR bundle

Parameter	*Accuracy*	*PDF*
Pressure	± 1 %	Normal
Flow Rate	± 1 %	Normal
Power	± 1.5 %	Normal
Inlet Temperature	± 1.5 C	Flat
Subchannel Area	± 0.5 %	Normal
Single-phase mixing coefficient	$2\sigma = \pm 42$ %	Normal
Two-phase multiplier of the mixing coefficient	$2\sigma = \pm 24$%	Normal
Equilibrium distribution weighing factor in void drift	$2\sigma = \pm 14$ %	Normal
Nucleate boiling heat transfer coefficient	$2\sigma = \pm 24$ %	Normal
Interfacial drag coefficient (bubbly flow)	$2\sigma = \pm 32$ %	Normal
Interfacial drag coefficient (droplet flow)	$2\sigma = \pm 26$%	Normal
Interfacial drag coefficient (film flow)	$2\sigma = \pm 36$ %	Normal

Source – PSU and GRS

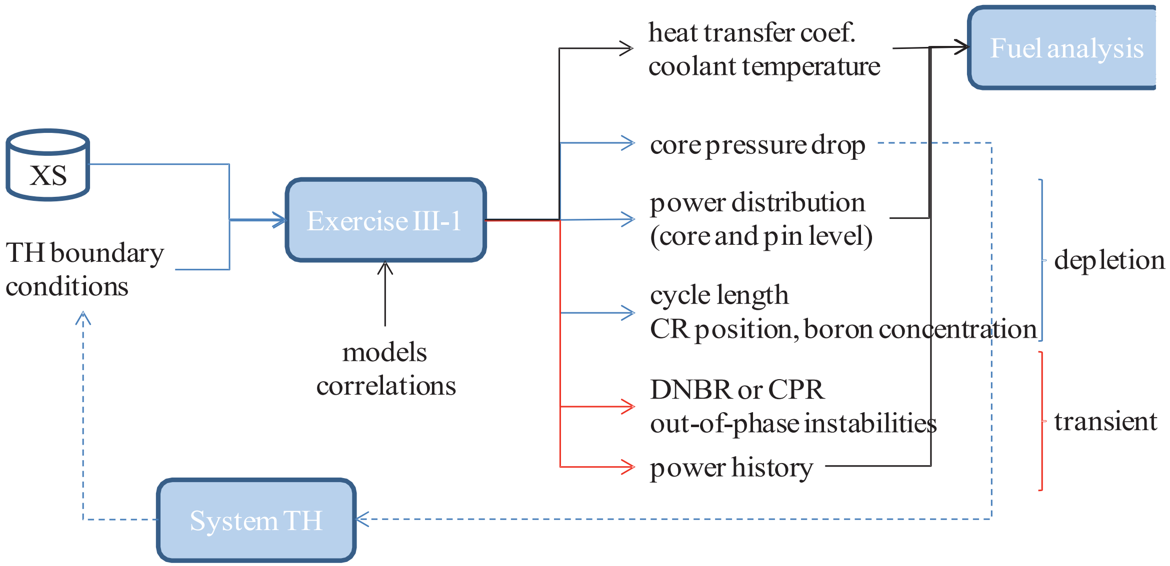
Phase III

Phase III - System Phase

- ✓ **Exercise III-1 – Core Multi-Physics:** Coupled neutronics/thermal-hydraulics core performance: U-7 (uncertainties in coupled history (depletion) and instantaneous feedback (transient) modeling)
- ✓ **Exercise III-2 – System Thermal-Hydraulics:** Thermal-hydraulics system performance: U-8 (uncertainties in thermal-hydraulics boundary conditions)
- ✓ **Exercise III-3 – Coupled Core/System:** Coupled neutronics kinetics thermal-hydraulic core/thermal-hydraulic system performance: U-9 (uncertainties in safety related parameters and margins)
- ✓ **Exercise III-4:** "Comparison of BEPU vs. Conservative Calculations"

Main input and output parameters in Exercise III-1

Source - SEA



Status of the Benchmark Activities

- Benchmark web-site: *http://www.nea.fr/html/science/egrsltb/UAM/index.html*
- Version 2.0 (final) of the Volume I of OECD LWR UAM Benchmark Specification (Phase I) has been finalized
- Version 1.0 (draft) of the Volume II of OECD LWR UAM Benchmark Specification (Phase II) is being finalized
- April 13-15, 2011 – UAM-5 workshop in 2011 in Stockholm, Sweden and was hosted by KTH – 52 participants from 27 organizations of 17 countries
- The SCALE 6.1 has been released in July 2011, which is important for some participants to perform Exercises I-2 and I-3
- The UAM-6 workshop is scheduled for May 9-11, 2012 in Karlsruhe, Germany hosted by the Karlsruhe Institute of Technology (KIT) - University of the State of Baden-Wuerttemberg and National Laboratory of the Helmholtz Association

Conclusions

- It is expected that the application of coupled codes for safety analyses will be continuously growing
- In fact, they are the only means to perform best-estimate calculations for accident conditions with a tight coupling of neutronics and thermal-hydraulics effects
- The current tendencies in coupled code developments are towards systematic integration of uncertainty and sensitivity analysis with simulations for safety analysis
- The OECD LWR UAM benchmark activity is designed to address current regulation needs and issues related to practical implementation of risk informed regulation
- Establishing such internationally accepted LWR UAM benchmark framework offers the possibility to accelerate the licensing process when using coupled best estimate methods

PREMIUM – Benchmark on the quantification of the uncertainty of the physical models in the system thermal-hydraulic codes

Tomasz Skorek

GRS Garching, Germany

Agnès de Crécy

CEA Grenoble, France

OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations,

Barcelona, Spain, 16-18 November 2011

PREMIUM = Post BEMUSE REflood Models Input Uncertainty Methods

⇒devoted to the uncertainties of the model related uncertain input parameters with a selected application case: the reflood prediction.

For the uncertainty methods of probabilistic type (« GRS type»)

- The BEMUSE benchmark,
- GRS investigation of ATHLET code uncertainty analyses
- Answers to the WGAMA questionnaire on the Use of Best-Estimate Methodologies

have clearly shown that improvements are necessary for the quantification of the uncertainty of the « input parameters ».

Recall: in probabilistic methods, these uncertainties are propagated through the considered code in order to obtain the uncertainty of the solution variables (output parameters).

Example of output parameters: Peak Cladding Temperatures.

Uncertain input parameters

Examples of input uncertainties:

- Initial and boundary conditions;
- Facility description/modelling;
- Material properties;
- Physical models;

Estimating the uncertainty of the physical models, e. g.: finding a probability distribution function of the multiplication factor for the model output.

It is:

- essential because model related input parameters are often among the most influential ones on the outputs;
- difficult because the majority of physical models outputs are not directly measurable.

Estimation of other input uncertainties, like initial and boundary conditions is more the question of availability and interpretation of the sources of information concerning initial and boundary conditions, facility description, etc.

Quantification of model uncertainties

Methods of physical model uncertainties quantification:

- Evaluation of separate effect tests
- Information obtained by model development, if available
- Experience from code validation
- Survey of expert state of knowledge
- Physical limitations
- Application of so called “intermediate” experiments (devoted to physical processes with a limited number of phenomena, e.g.: reflooding, critical discharge), where using sophisticated statistical methods uncertainties of physical models related to some phenomena can be obtained

Preferred way of quantification of physical models uncertainties – comparison with separate effect tests

For those phenomena for which separate effect tests do not exists application of “intermediate” experiments is advisable.

- Evaluation of the “intermediate” experiments is difficult and requires availability of sophisticated methodologies

Consequence: Expert judgment is too often used.

PREMIUM is aimed at solving this issue: the quantification of the physical model uncertainties

A particular case is considered: the physical models involved in prediction of core reflooding.

Why reflooding?

- Reflood is an important phenomenon for LB-LOCA, with a lot of possible modelling applications. At the end of the benchmark, the participants will have an estimation of the uncertainties of their code, to be considered for reflood prediction.
- Reflood experiments are of “intermediate” type: A limited number of physical models are involved: neither too complex (input uncertainties can be clearly identified), nor too simple case (for single effect tests model relevant uncertainties can be quantified by direct comparison with experimental data).

Examples of physical models by reflooding:

- Heat transfer downstream from the quench front;
- Enhancement to the heat transfer very close to the quench front;
- Relative velocities upstream or downstream from the quench front.

❖For some part of the phenomena model uncertainties can be quantified on the basis of separate effect tests

❖For other phenomena, e.g.: heat transfer enhancement close to the quench front, separate effect tests do not exist

Outline of the benchmark

PREMIUM goal: push forward the methods of physical models uncertainties quantification in thermal-hydraulics codes; in particular according to models describing phenomena for which no single effect tests exists

Coordination:

- The definition of the benchmark is the result of a joint effort of CEA and GRS.
- A coordination committee was created in April 2011 and comprises CEA, CSN, GRS, IRSN, UPC and UNIPI.

Schedule:

- Beginning of the benchmark: January 2012 (first meeting in February 2012)
- End of the benchmark: 2014.
- Possible extension: Preparation of "good practice guide" for model uncertainties quantification.

5 phases are defined, with a meeting for each of them, followed by the writing of a report

Benchmark specification

The general idea of the methods considered for the benchmark - Quantification of the uncertainty of the models using the measured data of reflood experiment in order to derive the uncertainties of the physical models involved in the reflood simulation for which separate effect tests do not exist and validation of the quantified input uncertainties performing uncertainty (and sensitivity) analyses of selected tests of another reflood experiment

Selected reflood experiments: FEBA/SEFLEX and PERICLES 2-D experiments.

- Availability of the measured data of these experiments has been checked
- Both tests cower similar field of application
- Using FEBA for the quantification of the uncertainties and PERICLES 2-D for the confirmation step follows the way of doing for a typical application:
  - simpler geometry for uncertainties quantification ;
  - application to larger scale facility like in the case of reactor application.

5 phases have been identified:

1. Introduction of the benchmark and methodology review
2. Identification of potentially important input uncertainties and they preliminary quantification;
3. Evaluation/Quantification of the uncertainties, by using the results of FEBA or equivalent experiment;
4. Confirmation/Validation step, calculation of selected PERICLES 2-D tests without knowing the test results;
5. Final synthesis report.

Phase I: Introduction of the benchmark and methodology review (coordinated by UPC)

- Detailed specification of the FEBA and PERICLES test facilities will be supplied to participants
- Presentation of "sophisticated" methods for quantification of input uncertainties on the basis of "intermediate" experiments (in this case reflood experiments) like CIRCE method (CEA) etc.
- The participants can:
  - either choose such a method;
  - or use simpler (conventional) approach, for instance trial-and-error method
- Presentation of test and facilities are going to be used by participants for input uncertainties quantification instead of FEBA experiment

- Kick off meeting: End of February 2012, Paris

Phase II: Identification of influential input uncertainties and they preliminary quantification (coordinated by Pisa University)

- Identification of important phenomena and related models
- The participants have to select the potentially important input uncertainties by reflood simulation according to the models applied in their codes. Reasons or rationales for selection should be given.
- Preliminary quantification of input uncertainties: for those models which can be quantified on the basis of separate effect tests, these tests should be applied.
- Definition of at least one common uncertain input parameter for all participants. (for example related to the heat transfer downstream from the quench front?):
  - If possible the same for all participants;
  - otherwise definition of common parameters for all users of the same code.
- ➢ Experimental data of FEBA experiment are to be distributed among the participants with Phase II specification

- Meeting: Begin of June 2012

Phase III: Evaluation/quantification of the model uncertainties using reflood experiment (coordinated by GRS)

Participants having in their disposal a tool for quantification of input uncertainties on the basis of “intermediate” experiments perform the quantification using:

- The results of their own experiment (SCTF, ACHILLES, RBHT, etc.) if:
  - The experiment is qualified enough;
  - They accept to make public their results.
- Otherwise they use experimental results of FEBA/SEFLEX experiment provided by GRS

Participants who will not apply such a tool for input uncertainties quantification

- The preferred way of quantification.
  - Quantification of model uncertainties on the basis of separate effect tests (for those phenomena/models for which such tests are available
  - Improvement of initially quantified input uncertainties on the basis of FEBA/SEFLEX experiments or equivalent using, e.g.: trial-and-error method to adjust uncertainty bounds and experimental results
- Quantification of input uncertainties in any other participant specific way with the aim to obtain consistence of uncertainty ranges with measured data for selected runs of applied flooding experiment

FEBA/SEFLEX Program

FEBA (Flooding Experiments with Blocked Array

8 test series have been performed under idealized reflood conditions

- Series I : Base line tests with undisturbed bundle geometry with 7 grid spacers
- Series II: Investigation of the effect of a grid spacer – with 6 grid spacers (without grid spacer at the bundle midpoint
- Series III to VIII: investigation of blockage effects – not suitable for the benchmark application

Fixed boundary conditions:

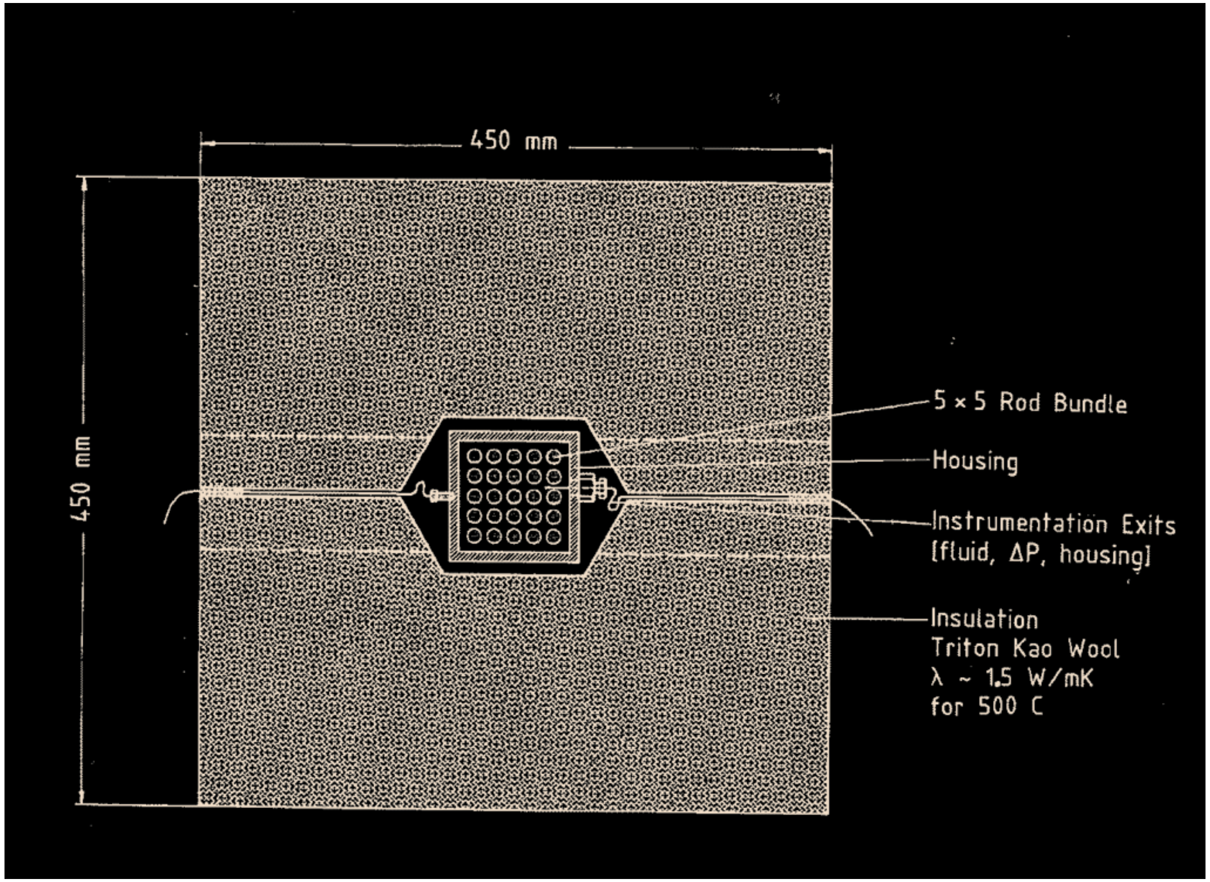
- Feed rates
- System pressure

Effect of reactor cooling system behaviour has not been taken into account

SEFLEX (fuel rod Simulator Effects in Flooding EXperiments

- The aim of the SEFLEX experiment: investigation of the influence of the rod simulator design and physical properties on heat transfer and quench front progression
- Series I - unblocked rod bundle: rods with helium filled gaps between Zircaloy claddings and aluminium pellets and 7 grid spacers
- Series II – unblocked bundle: rods with argon filled gaps between Zicaloy claddings and aluminium pellets and 7 grid spacers
- Series III and IV – rod bundle with blockage

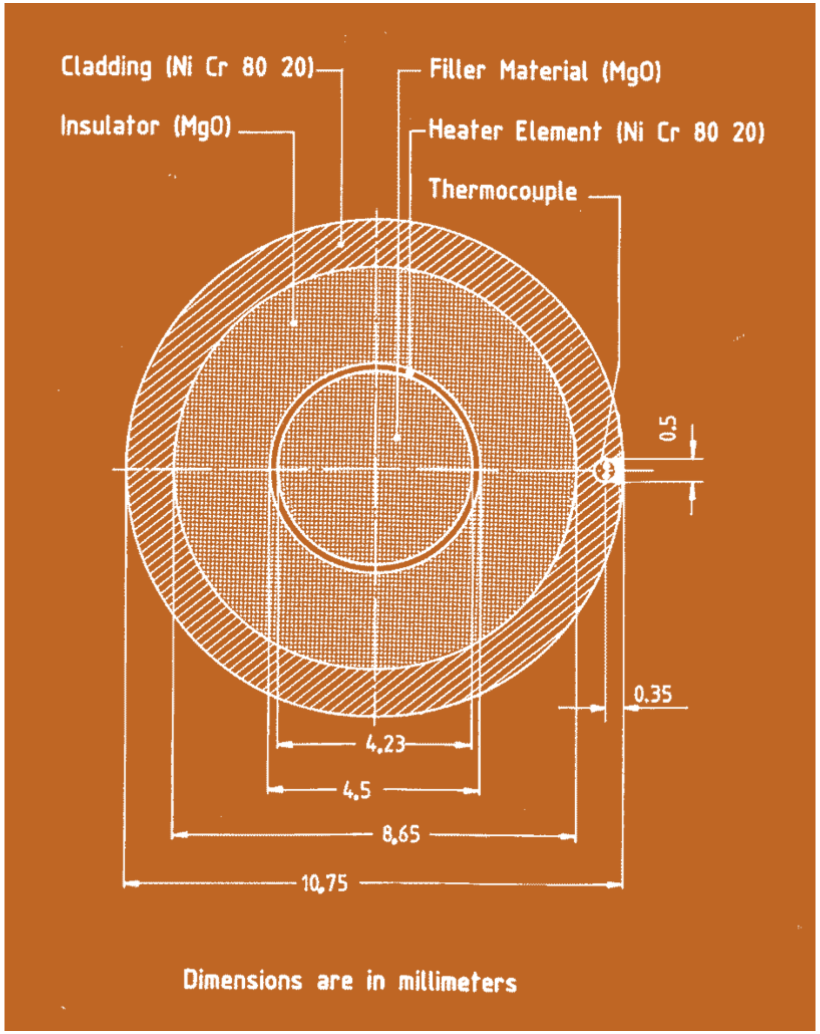
FEBA Test section

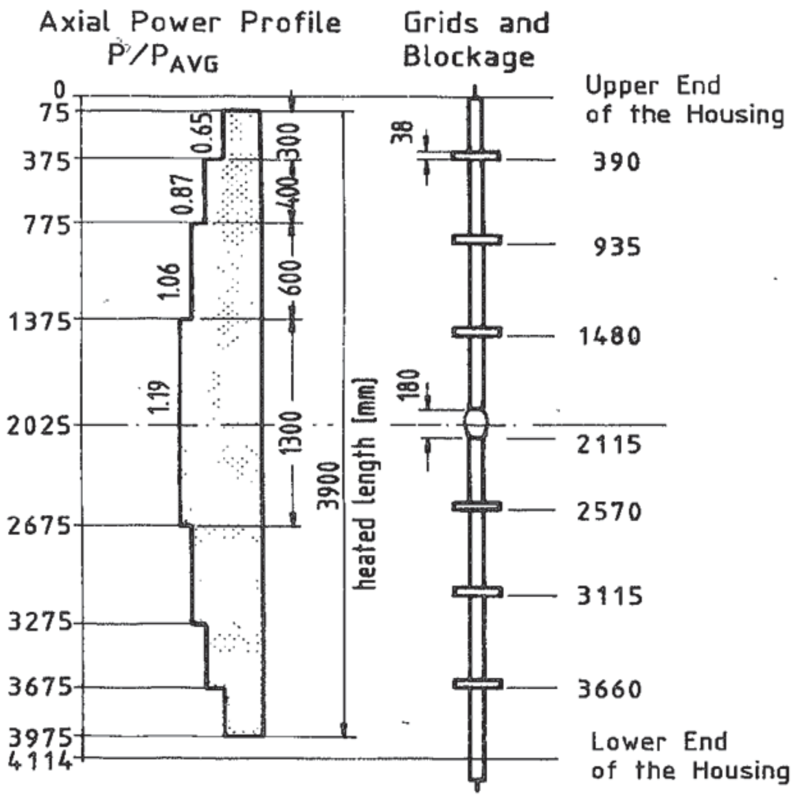


Cross sectional view of the test section with rod bundle

- 5 x 5 full length PWR fuel rod bundle
- Electrically heated rods with cosine axial power profile realized with 7 steps of different local power density
- Rod diameter – 10.75 mm
- Pitch: 14.3 mm
- Heated length: 3900 mm
- Hydraulic diameter: 13.47 mm – equal for all rods (housing so constructed that the peripheral rods have the same hydraulic diameter as inner rods
- Housing: 6.5 mm stainless steel

FEBA heater rod





FEBA Experiment – initial and boundary conditions

Series I

Test No.	Flooding velocity (cold), m/s	System pressure, bar	Feed water temperature, °C	
			0-30 s	End of test
223	3.8	2.2	44	36
216	3.8	4.1	48	37
220	3.8	6.2	49	37
218	5.8	2.1	42	37
214	5.8	4.1	45	37
222	5.8	6.2	43	36

Series II

Test No.	Flooding velocity (cold), m/s	System pressure, bar	Feed water temperature, °C	
			0-30 s	End of test
234	3.8	2.0	45	37
229	3.8	4.1	53	38
231	3.8	6.2	54	40
233	5.8	2.0	47	37
228	5.7	4.1	50	37
230	5.8	6.2	48	37

Bundle power:

- at the beginning of the test (0 s) : 200 kW
- during the transient: 120% ANS (measured value)

Phase III: Evaluation/quantification of the model uncertainties using reflood experiment (coordinated by GRS)

Experimental data of FEBA/SEFLEX experiment, which are useful for input uncertainties quantification and will be delivered to participants:

- Measured initial and boundary conditions: inlet velocity, feed water temperature, bundle power, system pressure, initial axial cladding profile
- Cladding temperature at 8 axial levels versus time
- Pressure drop measurements at lower middle and upper part of the test section
- Measured outlet conditions: water carry over, coolant temperature
- Fluid and housing temperature in the middle of the test section

- Only the tests without blockage are used for uncertainties quantification:
  - Series I and II of FEBA experiment
  - Series I and II of SEFLEX experiment

Phase III: Evaluation/quantification of the model uncertainties using reflood experiment (coordinated by GRS)

Results are to be obtained in the Phase III:

- Set of finally quantified input uncertainties
- Results of uncertainty analysis of one (or two) selected test run of FEBA or of the own reflood experiment considered by the participant in the Phase III:
  - Cladding temperature time trends
  - Time trend of pressure drop along the test section or water carry over the test section

Comparisons previewed are to be performed within the Phase III:

- Ranges of quantified uncertainty input parameters for the users of the same code
- Rough comparison of uncertainty analysis results for the selected FEBA test run (or equivalent experiment)
- Preliminary quantified ranges of input uncertainties with the final ranges obtained after evaluation during the Phase III

Phase IV: Confirmation/Validation of the input uncertainty ranges found in Phase III, by using PERICLES-2D results (coordinated by CEA and IRSN)

- Performing of uncertainty analyses of selected test runs from PERICLES experiment. The probability distribution functions/ranges of input uncertainties obtained in the previous step are to be applied.
- Comparison of calculated uncertainty ranges of selected output parameters (cladding temperature time trends and may be others) with corresponding measured values.

Interest of PERICLES-2D:

- The PERICLES experiments investigate the 2-D effects, among considered there are 2 tests with the same power for the 3 assemblies, i.e. without 2-D effects, and 3 tests with various power of hot and cold assemblies;
- The sequence quantification of input uncertainties on the basis of experiments on FEBA test facility and verification using PERICLES experiment follows a typical way of uncertainty analysis: input uncertainties are quantified on the basis of experiments and applied for reactor geometry of the much larger scale
- The selected tests from PERICLES experiment are proprietary tests, which have been given free for the purpose of the benchmark. Since, they have not been published up to now uncertainty analysis without knowing the test results are possible. It enables that more realistic validation can be performed

1. The PERICLES-2D experimental program

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Goal: Study the effect on the reflooding of the power difference between assemblies.

3 assemblies:
One hot assembly (B) surrounded by two cold assemblies (A and C). Their power is electrically supplied, by an independent way for each assembly.

- F_{xy} is the radial peaking factor, ranging from 1 to 1.85 (1 and 1.435 for the tests considered for PREMIUM).
- The axial power profile is of cosine type, with 11 steps:

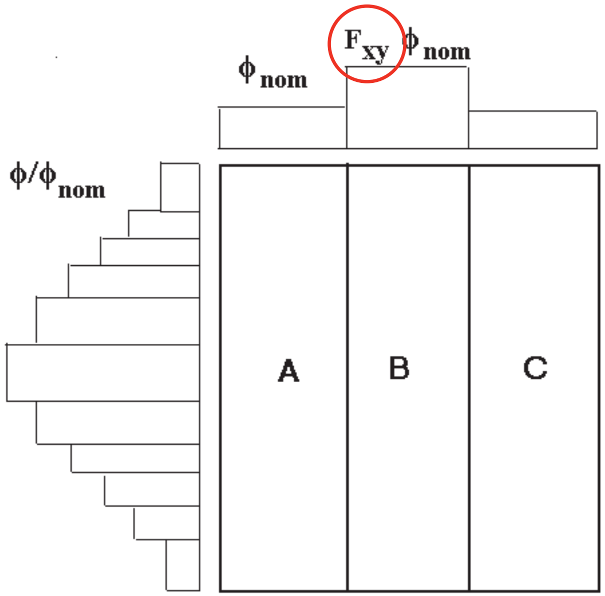

Fig. 3 Heating of the rods in the three assemblies

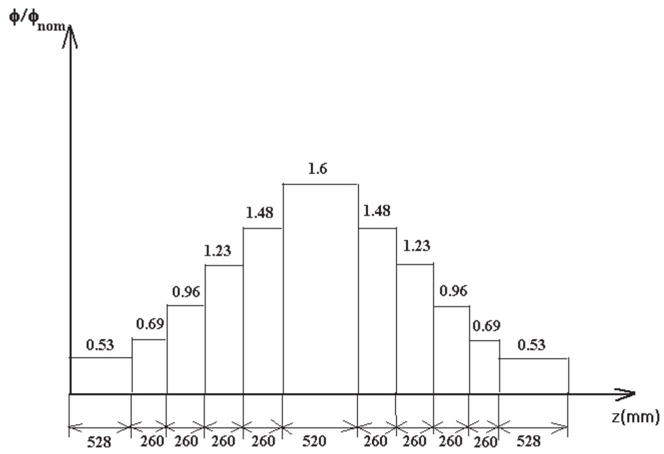

Fig. 4 Axial peaking factor versus elevation

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1. The PERICLES-2D experimental program

The assemblies are 17*7 rods, with full length (3.656 m).

⇒ Total number of fuel rods = 357

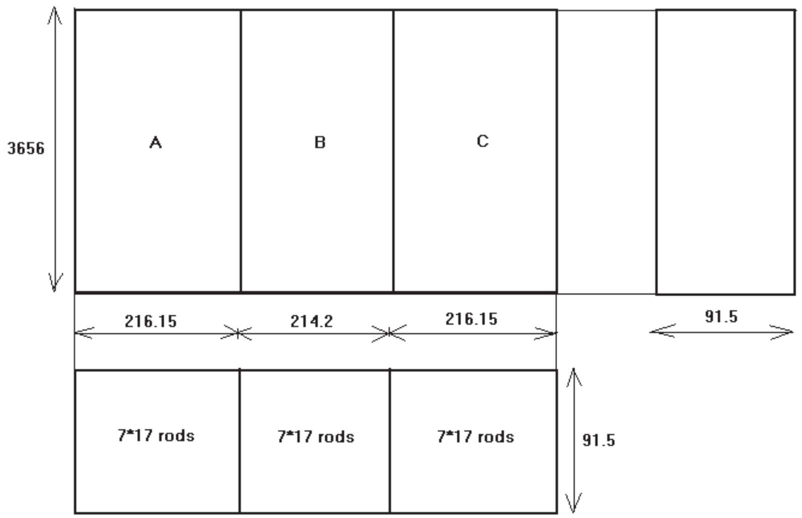


Fig. 1 The 2D PERICLES experiment (dimensions indicated in mm)

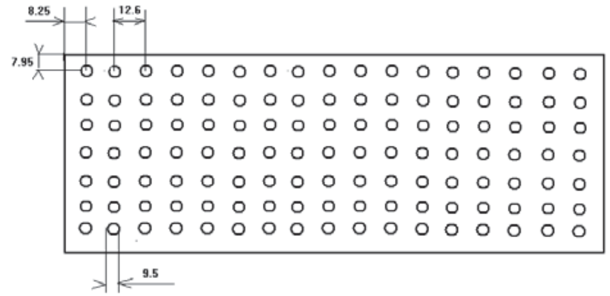
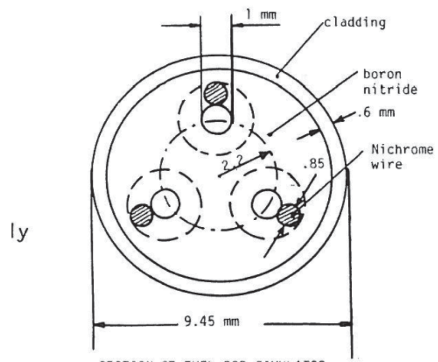


Fig. 2 Horizontal section of one assembly (dimensions indicated in mm)

8 spacer grids: z = 110, 668, 1180, 1691, 2223, 2748, 3298 and 3803 mm (≈500-550 mm between two spacer grids).

The simulators of fuel rods:



SECTION OF FUEL ROD SIMULATOR

- Cladding in stainless steel
- Insulator in boron nitride
- Heating element: 3 helical nichrome V wires

2. The tests considered for PREMIUM

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5 tests among roughly 40 tests

Test No	Φ_{nom} (HA) W/cm²	Φ_{nom} (CA) W/cm²	Fxy	GO (HA) g/cm²s	GO (CA) g/cm²s	T_{wi} (HA) °C	T_{wi} (CA) °C	DT °C	P (bar)
RE0062	2.93	2.93	1	3.6	3.6	600	600	60	3
RE0064	4.2	2.93	1.435	3.6	3.6	600	475	60	3
RE0069	2.93	2.93	1	3.6	3.6	475	475	60	3
RE0079	4.2	2.93	1.435	3.6	3.6	600	475	90	3
RE0080	4.2	2.93	1.435	5	5	600	475	60	3

- Φ_{nom}: nominal heat fluxes
- Fxy: radial peaking factor
- GO: inlet mass velocity
- T_{wi}: initial cladding temperature in the middle of each assembly, for which the injection is started
- DT: subcooling of the inlet water

- RE0064 test: reference test
- RE0069 and RE0062: effect of Fxy + initial cladding temperature T_{wi}
- RE0079: Effect of subcooling DT
- RE0080: Effect of inlet velocity GO

The validation step can be made by considering separately the tests with 2-D effects (Fxy = 1.435) and the tests without them (Fxy = 1).

2. The tests considered for PREMIUM

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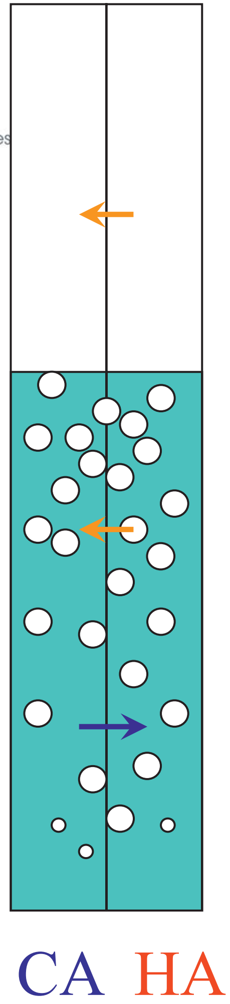
Some considerations about the 3-D effects



Main 3-D effects observed during reflood tests:

- Crossflow of Liquid from CA to HA assembly below the QF.
- Crossflow of Vapour from the HA to the CA below the QF and above it, in the “dry zone”.

These 3-D effects can be easily modelled using a 3-D code (TRACE, MARS, CATHARE) or a multi-axial + crossflows junctions modelling (RELAP5, ATHLET).

Phase V: PREMIUM conclusions (coordinated by CSN)

- Final report involving recommendation for quantification methodology of model uncertainties
- If desired, a "good praxis" guide of model uncertainties quantification can follow

End of the benchmark.