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Workshop Proceedings
Barcelona, Spain
16-18 November 2011

Part 4

OECD
BETTER POLICIES FOR BETTER LIVES

NEA
NUCLEAR ENERGY AGENCY

Organisation de Coopération et de Développement Économiques
Organisation for Economic Co-operation and Development **05-Dec-2013**

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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations

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Hosted by the Technical University of Catalonia (UPC) with support from the Spanish Nuclear Safety Council (CSN)

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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.

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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."

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

Use and application of “best estimate plus uncertainty” methods. A regulatory view

Fernando Pelayo & Rafael Mendizábal
Consejo de Seguridad Nuclear (Spain)

Barcelona, November 16-18, 2011

Use and application of "best estimate plus uncertainty" methods. A regulatory view

Focus of the presentation:

- To review the regulatory basis and development in support of BEPU methods,
- To emphasize what BEPU methods contribute to increase safety by quantifying uncertainty,
- To highlight some specific topics linked to BEPU methods licensability as possible lines of future development.

Structure:

- Regulatory analysis,
- Selected topics,
  - Deterministic safety analysis and probabilistic safety margin,
  - Probabilistic acceptance criteria,
  - Uncertainty methodologies,
  - BEPU methods and validation,
  - Characterisation of models uncertainty,
  - Compatibillity with Technical specifications,
  - Code accuracy and user qualification.

Use and application of "best estimate plus uncertainty" methods. A regulatory view

Regulatory analysis (1/6)

- Regulatory environment is characterized by its prevention to change.

The reasons of this prevention are manifold. It is in the benefit of the licensing process to have a stable and coherent regulation such that the expectations can be reasonably predicted both in resources and results

a well defined and stable regulatory framework is needed.

- Moving from a regulation that has proved to adequately fulfill its objectives like classic deterministic safety assessment to a new one like the Best Estimate Plus Uncertainty (BEPU), calls for further justification.

Use and application of "best estimate plus uncertainty" methods. A regulatory view

Regulatory analysis (2/6)

- All hazardous industries have in common the existence of <u>basic safety principles</u> developed in legally binding regulations to protect the affected patrimony (life, environment, property) as a result of the unwanted effects arising from accidents, use or any other means.
- Basic safety principles can be derived from applicable regulation in different countries with origin on international treaties or constitutional mandates
- Focusing on nuclear power industry, IAEA "Fundamental Safety Principles" SF-1 compiles and develops these safety principles for the case of civil nuclear activities.
- <u>Prevention</u> and <u>mitigation</u> from Nuclear Safety Convention, and <u>indemnification and reinstatement</u> from nuclear liability conventions constitute a set of core principles.

Use and application of “best estimate plus uncertainty” methods. A regulatory view

Regulatory analysis (3/6)

Precautionary principle an BEPU methods

- A precautionary action is defined (COM(2000)1) as: *“a decision exercised where scientific information is insufficient, inconclusive, or uncertain and where there are indications that the possible effects may be potentially dangerous and inconsistent with the chosen level of protection”*.
- Application of such principle calls for the acquisition of scientific knowledge in order to determine with sufficient certainty the risk in question, as well as observe how once the certainty level is quantified and risk reduction is feasible to a societal acceptable level a regulatory decision can be adopted.

BEPU methods contributes to fulfill the precautionary principle.

Use and application of "best estimate plus uncertainty" methods. A regulatory view

Regulatory analysis (4/6)

From classic DSA to BEPU

- The beginning:
  - Basically a theory of protection had to be built from scratch and concepts like defence in depth, redundancy, diversity, etc. were introduced in the design.
  - the corresponding accident analysis with the associated methodology of classification, acceptance criteria and analytical tools was developed.
  - Engineering works (early 70's) were heavily conditioned by the limited scientific and technology knowledge, limited data and computing power. As a result a common engineering practice was adopted making use of highly conservative and simplified assumptions resulting in the classic deterministic methodology and regulation.
  - For the case of USA regulations:10CFR 50 app.A (1971), R.G. 1.70 (1972), ANSI/ANS 18.2 (1973), 10CFR 50.46 app.K (1974) (interesting because of its prescriptive nature).

Use and application of "best estimate plus uncertainty" methods. A regulatory view

Regulatory analysis (5/6)

From classic DSA to BEPU

- The evolution:
  - Extensive experimental (LOFT, LOBI, LSTF,...) and scientific work (development of sophisticated fluid transport formalisms as well as suitable closure relations) was made, mainly after TMI, in order to gain data and knowledge and so to reduce the scientific uncertainty.
  - All that contributed to the development of a data base suitable for code validation and verification, resulting in the development of Thermal Hydraulic (TH) realistic system codes representing the best available knowledge.
  - It was early recognised that as a result of the imperfect knowledge of nature, as well as from the mathematical formalism reduction (averaging process, discretization, numerics, etc.) they produce uncertain results.
  - In order to fully exploit their potential and fulfill the objective of quantifying the uncertainty in their predictions their use should be linked to a sound uncertainty assessment (early recognised in NUREG/CR-5249, and NEA UMS benchmark).
  - New regulation was developed accordingly (10.CFR 50.46 (a)(1)(i)), R.G.1.157,...)

Use and application of "best estimate plus uncertainty" methods. A regulatory view

Regulatory analysis (6/6)

From classic DSA to BEPU

- The present:
  - ○ BEPU methods are penetrating into the licensing basis of NPP in numerous countries.
  - ○ Regulation is developing or being adopted elsewhere
  - ○ Numerous activities from IAEA and NEA in support of a sound development and application of BEPU methods.

As a summary of this part of the presentation, It's been discussed how BEPU methods better fulfils basic safety principles, in particular the precautionary. That the development of BEPU has a solid support in experimental and theoretical developments through an extensive V&V verification process. That regulation is currently allowing for its use.

But.... questions remain, which is part of the remaining of the presentation.

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Use and application of “best estimate plus uncertainty” methods. A regulatory view

DSA and PSM (1/4)

Benefits of using PSM

From a regulatory perspective BEPU methods have impacted the concept of safety margin.

- The use of BEPU methods derives into a definition of the safety margin based on likelihood measures, typically probabilities, thus allowing for a collective assessment of safety margins.
- The probabilistic safety margin (PSM) associated to a safety output V during a transient or accident sequence A with R_v the acceptance region is defined as:

$$PSM(V \mid A) \equiv PR(V \in R_V \mid A)$$

- Similarly and for a given initiating event (IE) with derived sequences $A_1,\ldots,A_s$, PSM definition can be extended to the IE

$$PSM(V \mid IE) = \sum_{j=1}^{S} p_j \, PSM(V \mid A_j)$$

Use and application of "best estimate plus uncertainty" methods. A regulatory view

DSA and PSM (2/4)

- Consider now that all the initiating events which induce transients of V could be grouped into M categories IE_i, i = 1, . . .M . The frequency of V exceeding the limits is:

$$\nu\left(V \in R_V^*\right)=\sum_{i=1}^{M}\nu_i\left[1-PSM\left(V \mid IE_i\right)\right]$$

and,

$$1-PSM\left(V \mid IE_i\right)=\sum_{j=1}^{S_j} p_{ij}\left[1-PSM\left(V \mid A_{ij}\right)\right]$$

We conclude that probabilistic safety margins can combine with initiator frequencies and produce exceedance frequencies, which constitute the plant risk.

Use and application of “best estimate plus uncertainty” methods. A regulatory view

DSA and PSM (3/4)

Classic DSA methodologies require of a BEPU method for validation

- PSM can be expressed in terms of the probability distributions of both V and L:

$$PSM(V \mid IE) = \int_{-\infty}^{+\infty} f_L(s) F_V(s \mid IE)\, ds$$

- Let us now suppose that V_b is a large value of V obtained from a limiting scenario evolving from IE and calculated with a conservative methodology. Typically, V_b would correspond to a Design Basis Transient (DBT). This means that the probability of V being less than V_b conditioned to IE is very close to 1. from above equation derives the following inequality:

$$PSM(V \mid IE) > PSM(V \mid DBT) \cdot PR(V \le V_b \mid IE)$$

with, $$PSM(V \mid DBT) \equiv PR(V_b < L \mid IE)$$

- which is the PSM for the design basis transient.

DSA and PSM (4/4)

Classic DSA methodologies require of a BEPU method for validation

- Considering M categories of initiating events we obtain an upper bound for the exceedance frequency of L

$$\nu(V > L) < \sum_{k=1}^{M} \nu_k PR(V \le V_{bk}) PR(V_{bk} \ge L) + \sum_{k=1}^{M} \nu_k PR(V > V_{bk})$$

- The right hand side gives the exceedance frequency of L supposing that the BDBS result in a limit violation (i.e. progress beyond design basis assumptions).

- If the DBT is adequately chosen, the residual term can be neglected against the main one, and the maintenance of the safety margin is assured through the enveloping character and the safety margin of the DBTs. This is the basis of the deterministic design approach.

- Above equation reduces to the classical fully deterministic approach when $PR(V_{bk} \ge L) = 0$ and $PR(V \le V_{bk}) = 1$ (a strong statement that calls for proof) and as a result the frequency of exceedance for a given IE is that of the BDBS which is the PSM for the design basis transient.

- Traditional safety margin, defined as L minus V_b may be misleading as it carries no information on the probability to exceed the safety limit, so it needs to be supplemented by a validation against a BEPU method. This in fact has been a driving force for the development of BEPU methods, and justifies a regulatory move.

Use and application of "best estimate plus uncertainty" methods. A regulatory view

Probabilistic acceptance criteria (1/2)

Damage limits should be imposed on residuals exceeding acceptance criteria

- For BEPU analyses, generally a non-zero probability exists to surpass the safety limit and as a consequence an escalation in the damage to happen. In order to estipulate a limit to the PSM it is basic to constrain the allowable damage for the residual cases that under a BEPU method will violate the limit.

High statistical confidence should be imposed based on limited size of output samples

- An acceptance criterion for the PSM of the enveloping transient should read:

$$PR\left\{PSM\left(V \mid DBT\right) > M_0\right\} \geq 1-\alpha$$

- The requirement of high statistical confidence is due to the finite size of the random samples used in the calculation of the PSM. The values of both M_0 and α are imposed by the regulatory authority. Typically M_0 and $1-\alpha$ have been both set to 0.95

Use and application of “best estimate plus uncertainty” methods. A regulatory view

Probabilistic acceptance criteria (2/2)

PSM by a pure Monte Carlo procedure can be envisaged as a computational binomial experiment

- Each Monte Carlo run has two possible outcomes: “success” if V falls inside the acceptance region, and “fail” otherwise. This is captured by a random variable S taking on the value 1 for success and 0 for fail. S is a Bernoulli random variable, with parameter PSM(V|DBT)

$$PSM\left(V \mid DBT\right) = E\left[S\right]$$

- The acceptance criterion may be multidimensional.

Use and application of "best estimate plus uncertainty" methods. A regulatory view

Uncertainty methodologies (1/2)

Ample freedom in uncertainty methods

- When uncertainty is taken into account, DSA acceptance criteria adopt the probabilistic form stated previously. There is no regulatory prescription about the methodology to use in the check of such criteria.

- There are two basic procedures for checking the acceptance criteria from a random sample of the output V
  - Construction of a tolerance region for V, with coverage/confidence levels of at least $M_0/(1-\alpha)$, and check that such region is inside the acceptance region R_V.
  - Construction of a lower confidence limit, with level $1-\alpha$, and check that it is higher than M_0

- A lower confidence limit with level $1-\alpha$ for the PSM is a statistic P_L such that:

$$PR\left\{P_L \leq PSM\left(V \mid DBT\right)\right\} = 1-\alpha$$

- Methods to set up confidence limits on a binomial probability (as PSM) can be either parametric or nonparametric. Nonparametric (i.e. distribution-free) methods can be classified as frequentist (or classical) and Bayesian.

Frequentist Clopper Pearson $P_L =_{\alpha} beta\left(R, N-R+1\right)$

Bayesian $P_L =_{\alpha} beta\left(\gamma + R, \delta + N - R\right)$

Use and application of “best estimate plus uncertainty” methods. A regulatory view

Uncertainty methodologies (2/2)

Ample freedom in uncertainty methods

- Parametric methods can also be used in the confidence limit estimation, when parametric distributions are assumed for V and L. Goodness-of-fit tests must be applied to the data, before confidently assigning the distributions.

- When the expected value of the PSM is quite large (i.e. very close to 1), it is clear that the acceptance limits are located in the tails of the V distribution. In such cases, there are sampling procedures to check PSM more efficiently than simple random sample, mainly stratified sampling and importance sampling, which focus on the tail regions of V close to the acceptance limits.

Use and application of “best estimate plus uncertainty” methods. A regulatory view

BEPU methods and validation (1/1)

Need for BEPU method validation

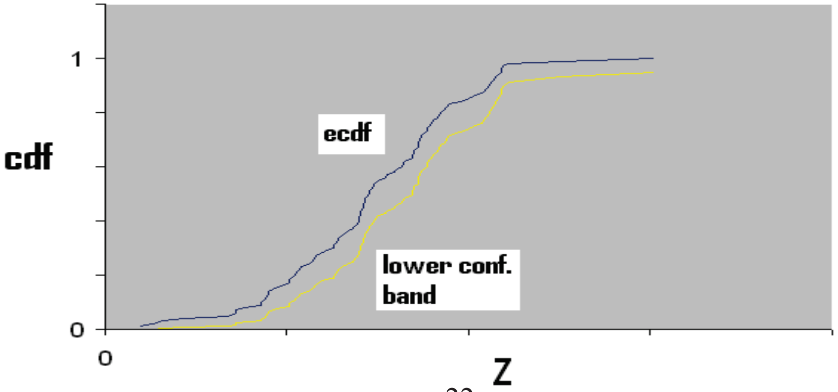
- BEPU methodologies should undergo a verification and validation process before being accepted for the licensing task. In the validation process, real data, from experimental facilities and plants, are compared with predictions of the methodology.
- The validation of realistic evaluation model is a different exercise from both the licensing and the conservative validation. Now the acceptance criteria are focused on the closeness of the calculated values of the important outputs to their real values, taking into account both prediction and measurement uncertainties. These important outputs can be different physical magnitudes, or magnitudes in different spatial locations and/or in different time instants or the actual safety limit variables.
- The relevance of Integral Effect Tests (IET) in validation must be stressed. They were formerly planned as a means to qualify the predictive capability and accuracy of codes , but they also play a fundamental role in the validation of uncertainty methodologies itself.
- IET have as peculiarity the controlled environment in which the experiments are performed. This means that uncertainties about initial and boundary conditions are reduced with regard to those from a NPP, whereas the model uncertainties remain.
- There are many methods for performing this validation. When we have a Monte Carlo random sample of the outputs, a possibility is to formulate the validation exercise as a statistical hypothesis test.

Use and application of “best estimate plus uncertainty” methods. A regulatory view

Characterisation of models uncertainty

Need for degree of conservatism

- Models uncertainty is characterised as epistemic, with origin in the limited knowledge and in principle reducible.
- PREMIUM NEA exercise aims at benchmarking methods to infer model pdf from codes output (inverse problem)
- The use of completely “realistic” pdf for input variables/parameters is questioned. Pdf should be assigned with some degree of conservatism.
- When probability distributions are assigned to model parameters from sample of experimental values, the statistical uncertainty, because of the limited size of the sample of experimental values, should be taken into account.



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Use and application of "best estimate plus uncertainty" methods. A regulatory view

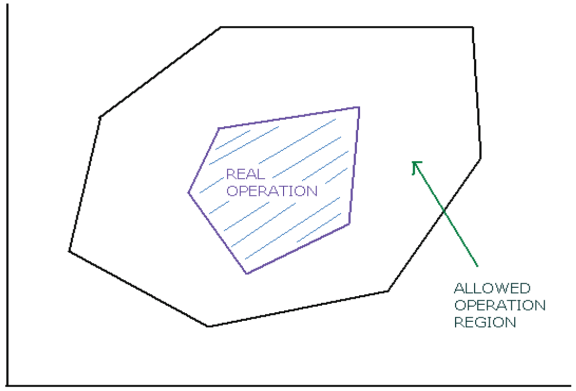
Compatibility with Technical Specifications (1/2)

Need for adapted pdf

- Nuclear Safety Convention requires (art.19.ii) *"operational limits and conditions derived from the safety analysis, tests and operational experience are defined and revised as necessary for identifying* *__safe boundaries for operation."__*

- Current interpretation is compatible with classic DSA, where LCO values of variables or parameters of a Design Basis Transient (DBT) are used, that is, DBT envelops extreme values of the operation space.

- BEPU methodologies based on pure Monte Carlo propagation (i.e. simple random sampling of the inputs) with a small or medium number of runs will not provide a real exploration of the regions close to the LCOs (unlikely but allowed).

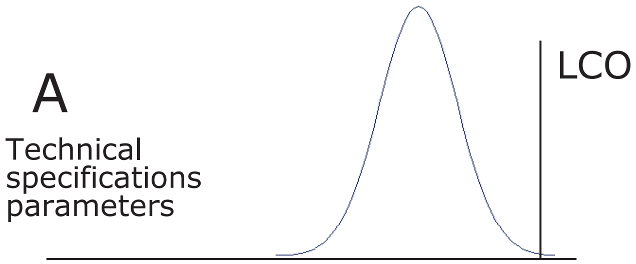


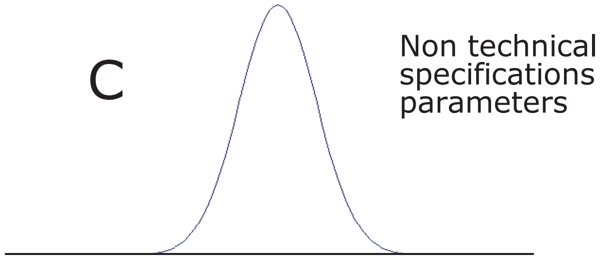
Use and application of "best estimate plus uncertainty" methods. A regulatory view

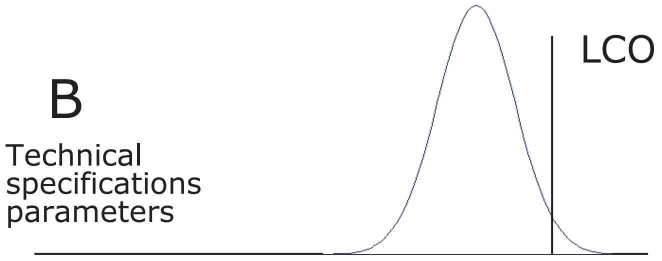
Compatibility with Technical Specifications (2/2)

Need for adapted pdf

- The crucial point is that safety analyses must prove not only that the *real* operation is safe, but also that the *allowed* operation of the plant is safe.
- Review of basic TecSpec concepts in view of BEPU methods is necessary







- A + C = validation of expected operation
- B + C = validation of LCOs

Use and application of "best estimate plus uncertainty" methods. A regulatory view

Code accuracy and user qualification

A must

- Prior to the development of an uncertainty method, code eligibility should be performed and accuracy determined through a verification and validation process.
- Code accuracy can only be claimed if confirmed through an exhaustive contrast against experimental, operational data and even analytical solutions.
- Code accuracy should be checked, and preserved, through scale variations.
- Training of the individuals, technology competence and resources are basic. For the case of licensing calculations, strict design procedures should be required.

CONCLUSIONS

- Best Estimate Plus Uncertainty (BEPU) methods are the result of a coherent effort from the nuclear safety technology community and fully consistent with regulation and basic principles.

- The move from classic DSA to BEPU methods implied the adoption of probabilistic acceptance criteria, and the definition of probabilistic safety margins (PSM)

- The probabilistic definition of safety margin allows the use of combination rules derived from probability theory.

- BEPU methods rigorously allow for the confirmation of the conservative nature of classic deterministic methods.

- A variety of uncertainty treatment methods possibilities exist which are in principle compatible with regulation.

- Validation of BEPU methods against Integral Effect Tests (IET) is needed. Development of specific methods to deal with that would be desirable.

- Special care should be exercised when developing probability distributions for model parameters. OECD PREMIUM benchmark will hopefully test different methods to deal with the issue.

- Implementation of BEPU methods with licensing purposes should appropriately consider its compatibility with Technical Specifications of the nuclear power plant.

- Code accuracy should be demonstrated to be scale independent for licensing scenarios where the BEPU method is intended to be applied.

- Strict rules with regard to code user qualification and design procedures should be part of the methodology assessment process.

THANK YOU

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Uncertainty Methods Framework Development for the TRACE Thermal-Hydraulics Code by the U.S.NRC

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Uncertainty Method Considerations

- TRACE is a “best-estimate” thermal-hydraulics code used by the NRC for audit calculations and resolution of safety issues for all light water reactors. Currently, TRACE “realistic” calculations, without uncertainties are used. Most recent code development has been focused on new and advanced LWRs (AP1000, EPR, ESBWR, etc.).

- We expect the industry to continue the trend towards Best Estimate (BE plus uncertainties), as opposed to conservative Appendix K analyses. A comparison between TRACE and industry calculations would be improved if the comparison were between 95/95 estimates (95% probability level at 95% confidence level).

- Eventually we expect BE Methods to be applied for more than just LOCA analysis (RIA, ATWS, BWR Stability, MSLB, Containment, etc.) .

- Gas cooled reactors will be analyzed with MELCOR/PARCS; a Best Estimate approach has been proposed. Uncertainty Methodology for GCRs is in the planning and design stages.

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TRACE Development Effort

- TRACE code development was initiated in about 2000 with the objective of consolidating the major NRC codes; TRAC-P, TRAC-B, RELAP, RAMONA.

- Significant efforts (2002-2009) aimed at extension of TRACE to new and advanced LWRs; AP1000, ESBWR, EPR, ACR-700. Some recent work (2009-present) is directed towards ABWR, SMR.

- Additional efforts (2010-present) on development and assessment expended on BWR ATWS and stability.

- Relationship to (NRC) Code Uncertainty
  - ***Generality in plant types***
  - ***Extension to new analyses***

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Framework Considerations

Framework Decisions Based On:

- Regulatory Acceptability
- Feasibility of Implementation
- Extensibility of the Methodology
- Ease of Use & Effort to Perform Analyses
- Resources to Implement Methodology

Evaluation of available methods lead to the conclusion that an “ordered statistics” approach should be used.

Framework Considerations

- NRC Regulations have added "uncertainty" to the BEPU process. The 1988 rule, for example states:

"when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of the section, there is a high level of probability that the criteria would not be exceeded."

- Reg Guide 1.157 suggests that a 95% probability is adequate - - however, this is a guide (not a regulation) and confidence level is not addressed.
- Uncertainty and acceptable uncertainty methods were likewise vague in the CSAU demonstration study. Uncertainty was determined using response surface methodology, but objective was not the statistical approach.

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Symbolic Nuclear Analysis Package (SNAP)

- Automate the sampling and ranging process using the SNAP uncertainty interface. SNAP is used to develop, modify and submit input decks. It currently has capability to automatically sample and change only input parameters, but modifications to both TRACE and SNAP are planned to accommodate the ranging of models and correlations.

- Plan is to use SNAP as the “front-end” in sampling process and use it to automatically modify and submit calculations as part of an uncertainty method. SNAP will also be used for the “back-end” evaluation of results.

- Being built in to SNAP are parameters to be sampled for a given plant type, the set of parameters to be ranged for a given scenario, and the range/distribution for those parameters. Users will have the option to modify recommended sets if necessary.

Approach: Uncertainty Parameters

- Available PIRTs (Phenomena Identification and Ranking Tables) have been reviewed for each plant type and accident scenario.

Plant Types:

- W 3- and 4-Loop PWR
- CE PWR
- B&W PWR
- GE BWR

- AP1000
- EPR
- APWR
- ESBWR
- ABWR

Transient Types:

- LBLOCA
- SBLOCA
- MSLB

Uncertainty Contributor Types:

☐ TRACE Code Model Parameters
☐ Input Model Parameters

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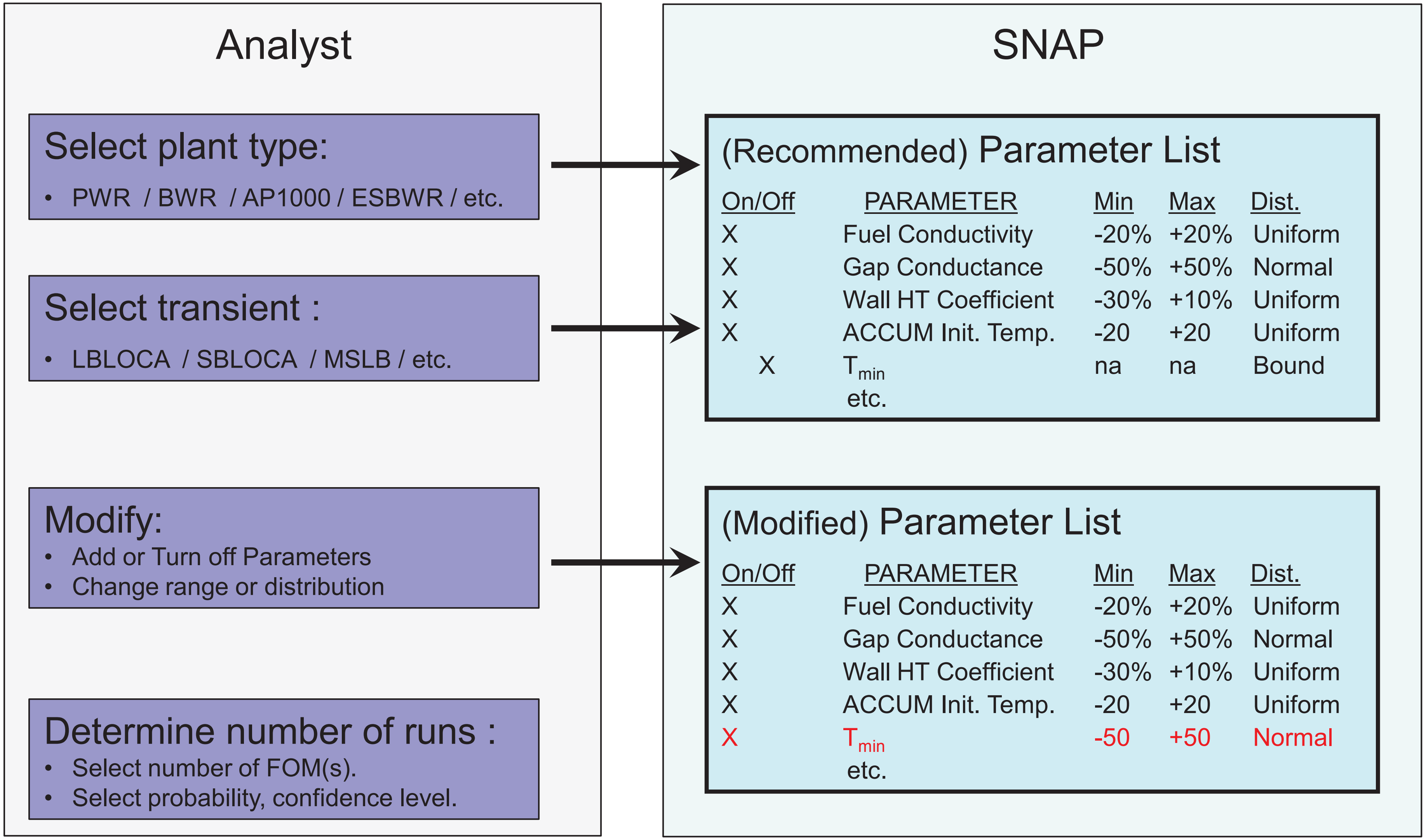
Approach: Analyst Actions



Analyst

Select plant type:
- PWR / BWR / AP1000 / ESBWR / etc.

Select transient :
- LBLOCA / SBLOCA / MSLB / etc.

Modify:
- Add or Turn off Parameters
- Change range or distribution

Determine number of runs :
- Select number of FOM(s).
- Select probability, confidence level.

SNAP

(Recommended) Parameter List

On/Off	PARAMETER	Min	Max	Dist.
X	Fuel Conductivity	-20%	+20%	Uniform
X	Gap Conductance	-50%	+50%	Normal
X	Wall HT Coefficient	-30%	+10%	Uniform
X	ACCUM Init. Temp.	-20	+20	Uniform
X	T_{min}	na	na	Bound
	etc.			

(Modified) Parameter List

On/Off	PARAMETER	Min	Max	Dist.
X	Fuel Conductivity	-20%	+20%	Uniform
X	Gap Conductance	-50%	+50%	Normal
X	Wall HT Coefficient	-30%	+10%	Uniform
X	ACCUM Init. Temp.	-20	+20	Uniform
X	T_{min}	-50	+50	Normal
	etc.			

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Figures of Merit (FOM)

- Conventional LOCA analysis has three FOMs; Peak Cladding Temperature (PCT), Maximum Local Oxidation (MLO), and Core-Wide Oxidation (CWO).

- Advanced LWRs may have different limits; Minimum Inner Vessel Mixture Level, Minimum DNBR, Maximum Containment Pressure, etc.

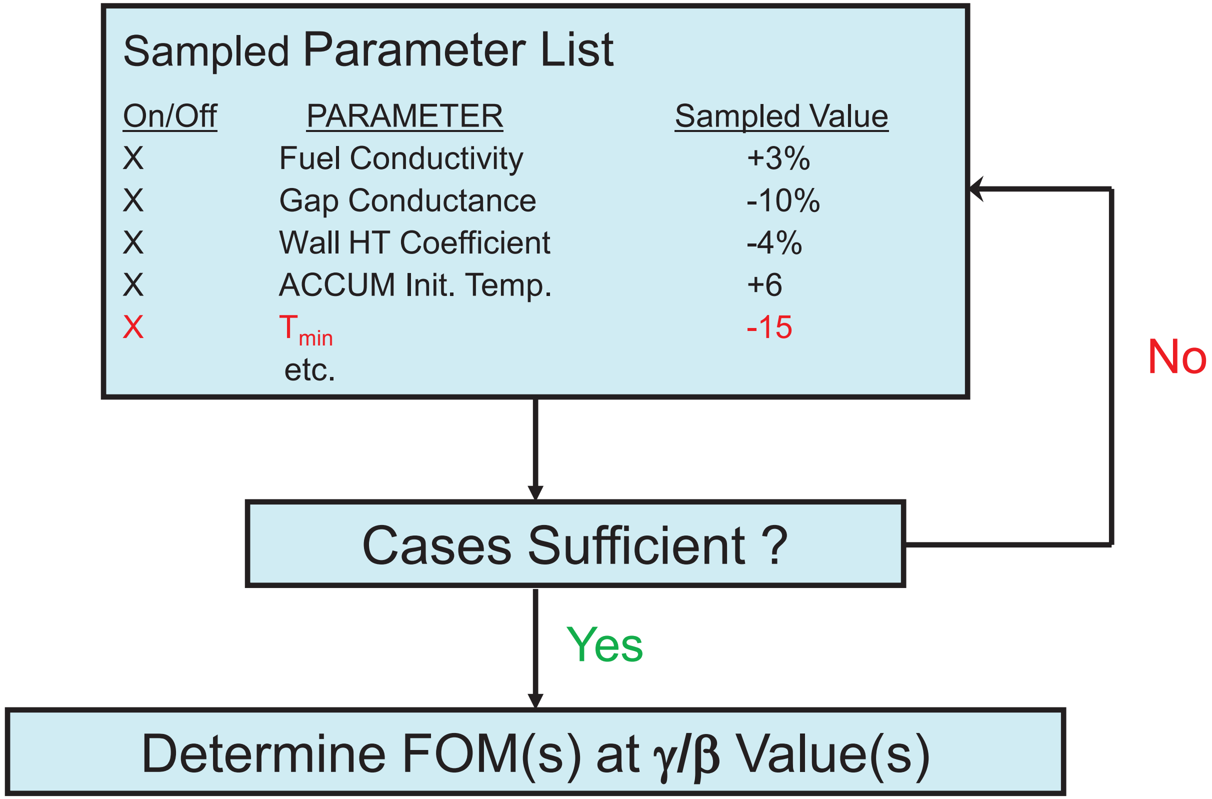
- TRACE will keep track of each FOM, and in some cases multiple FOMs must meet regulatory limits. Because of multiple FOMs, Guba-Makai-Pal will be used to determine the necessary number of calculations:

$$\beta = \sum_{j=0}^{N-p} \frac{N}{(N-j)!\,j!} \gamma^j (1-\gamma)^{N-j}$$

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Approach: Calculations

SNAP



Sampled Parameter List

On/Off	PARAMETER	Sampled Value
X	Fuel Conductivity	+3%
X	Gap Conductance	-10%
X	Wall HT Coefficient	-4%
X	ACCUM Init. Temp.	+6
X	T_{min}	-15
	etc.	

Cases Sufficient ?

No

Yes

Determine FOM(s) at γ/β Value(s)

DAKOTA

- The Design and Analysis Toolkit for Optimization and Terascale Applications (DAKOTA) is a code developed by SNL for sensitivity and uncertainty analysis.

- DAKOTA permits imposition of uncertainty on code inputs and performs post-processing for sensitivity and uncertainty calculations.

- DAKOTA is capable of generating samples (either random or Latin hypercube design) from a variety of distributions.

- Sensitivity methods include:
  - Morris One-At-Time (OAT)
  - Variance based approaches
  - Correlation (both Pearson and Spearman)
  - Local gradients

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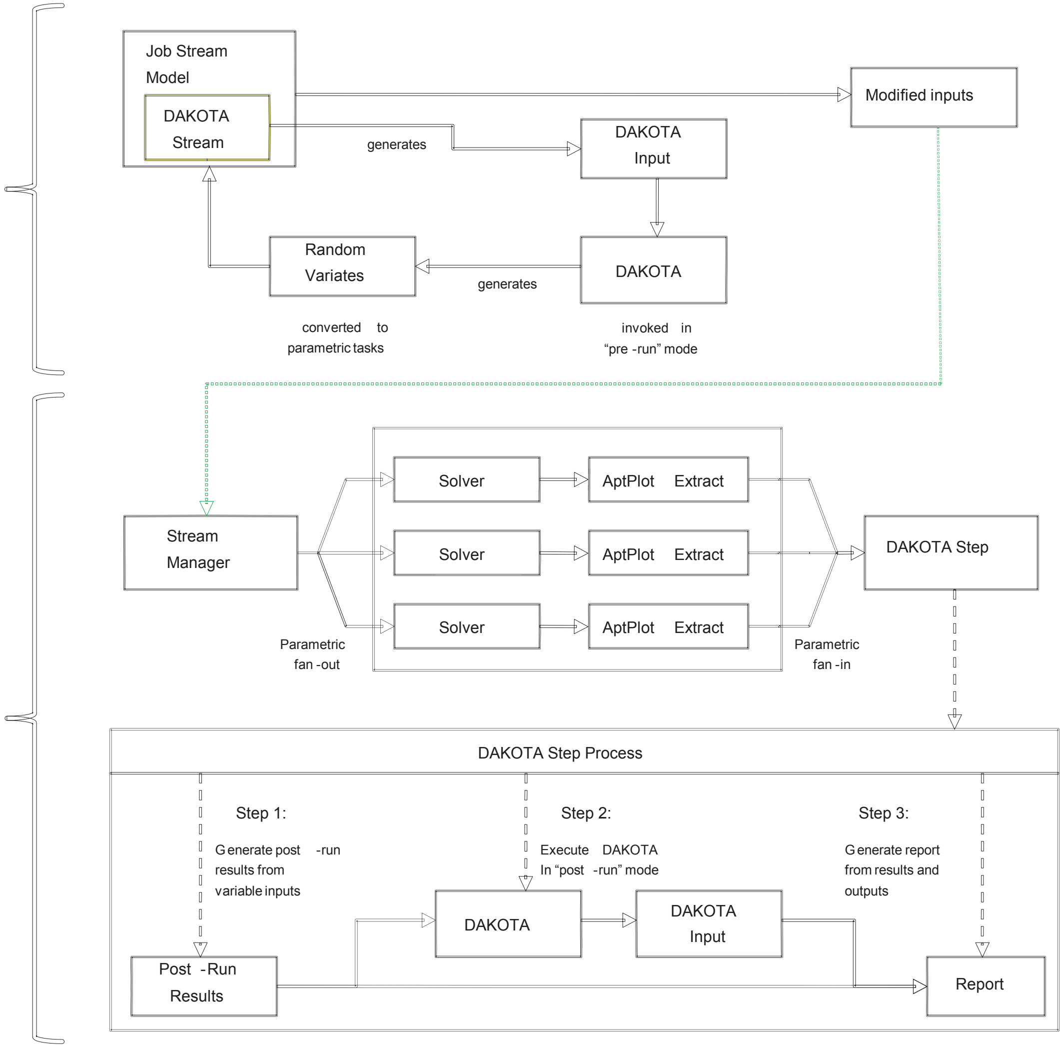
SNAP – DAKOTA Communication

Client -side:
Generates the uncertainty parametric models

Server -side:
Submitted Stream



Job Stream Model
DAKOTA Stream
Modified inputs
generates
DAKOTA Input
DAKOTA
generates
Random Variates
converted to parametric tasks
invoked in "pre -run" mode

Stream Manager
Solver
Solver
Solver
AptPlot Extract
AptPlot Extract
AptPlot Extract
Parametric fan -out
Parametric fan -in
DAKOTA Step

DAKOTA Step Process
Step 1:
Generate post -run results from variable inputs
Step 2:
Execute DAKOTA In "post -run" mode
Step 3:
Generate report from results and outputs
Post -Run Results
DAKOTA
DAKOTA Input
Report

Current Status

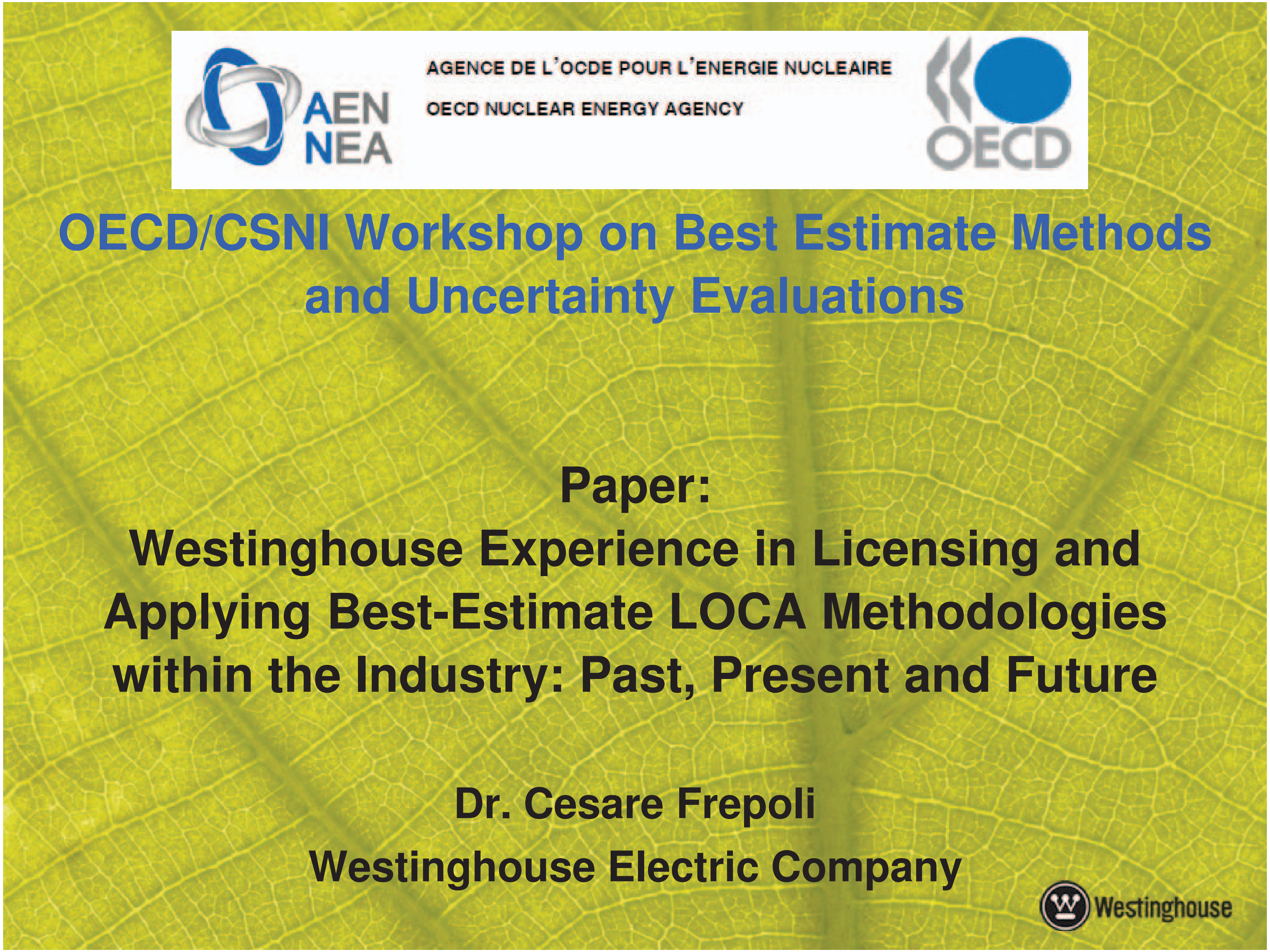
- Completed:
  - Automatic Calculation of “N” TRACE Calculations
  - Implementation of DAKOTA distribution types
  - Results Reporting
  - Uncertainty Analysis Templates
- Expected Soon:
  - Additional sensitivity analysis options
  - TRACE model & correlation parameter ranging

Current Status

- Addressing the "User Effect"
  - Previous investigations have found the so-called "User Effect" to be important.
  - To reduce (but not eliminate the effect), specific guidelines are being produced for TRACE applications.
  - Guidelines for LWR applications are integral with assessment for separate and integral effects tests.

Current Status

- Addressing model and correlation uncertainties:
  - Considered the most difficult challenge!
  - Performance many correlations are difficult to isolate in IETs or SETs.
  - In some cases, “bounding” a poorly understood phenomena is acceptable to both regulator and the applicant (but not maybe the statistician).
  - Effect of some models & correlations is temporal.



AGENCE DE L'OCDE POUR L'ENERGIE NUCLEAIRE

OECD NUCLEAR ENERGY AGENCY

OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations

Paper:

Westinghouse Experience in Licensing and Applying Best-Estimate LOCA Methodologies within the Industry: Past, Present and Future

Dr. Cesare Frepoli
Westinghouse Electric Company

Outline

- Industry and regulatory drivers for improved realistic (Best-Estimate) safety analysis methods
- Historical overview Westinghouse LOCA Safety Analysis Methodologies
  - Introduction to the new Westinghouse Full Spectrum LOCA Evaluation Model.
- Best-Estimate Plus Uncertainty (BEPU) within a Statistical Framework and its challenges
- Conclusions

Why BEPU LOCA Methods? Drivers for BELOCA

- Economic
  - Plant power uprates (more than 50% of BELOCA applications have been tied to uprate programs)
  - More efficient core designs (Increased peaking factor limits, reduced leakage)
  - Longer fuel cycles (up to 24 months)
- Equipment degradation
  - Increased steam generator tube plugging
  - Degraded safety injection flow
- Operational flexibility
  - Relaxed diesel generator start times
  - Relaxed Technical Specification limits
- Regulatory

History of Westinghouse Best Estimate LOCA Methodology, key events

- Licensed the 1st NRC-Approved (SER) Best-Estimate Large Break LOCA (BELOCA) Methodology in 1996 (CQD)
- Licensed an Automated Statistical Treatment of Uncertainty Method (ASTRUM, 2004)
- Extensive application of CQD (1996-2002) and ASTRUM (2002-2011) to most of PWR fleet for LBLOCA BEPU Safety Analysis
- The Full Spectrum LOCA Methodology now extends applicability to any LOCA: Large, Intermediate and Small Break scenarios
  - Development completed in 2010 and currently under USNRC review

FULL SPECTRUM LOCA

Westinghouse

Westinghouse BEPU Methodologies WCOBRA/TRAC Development History

Year	Development Activity
1983	Obtained COBRA/TRAC (NUREG/CR-3046) for UPI plant appl.
1988	SER on SECY-UPI method (Interim BE method/SECY-83-472)
1988	RG 1.157 “Best Estimate Calculation of ECCS Performance”
1989	CSAU (NUREG/CR-5249)
→1993	Improvements and Error corrections to COBRA/TRAC and Development of WCOBRA/TRAC through > 100 SET/IET test simulations
1993	Submittal of Code Qualification Documents (CQD) to USNRC
1996	SER by USNRC for the BELOCA methodology (1996 CQD Method)
1998	SER for DVI Plant (AP600)
1999	SER for UPI-BELOCA
2004	AP1000, SER for ASTRUM
2005-2010	Development of Full Spectrum LOCA (FSLOCA) Methodology and upgrade to WCOBRA/TRAC-TF2
2013	Expected date for FSLOCA SER

Current (Licensed) Westinghouse LOCA Technology

Large Break LOCA

- Best-Estimate Methods Based on WCOBRA/TRAC (PD2)
  - ASTRUM (2004), CQD (1996)
    - Valid from 1 ft^2 (0.093 m^2) to DEG (2 * 4.12 ft^2 (0.383 m^2))

Small Break LOCA

- Appendix K Methods Based on NOTRUMP (1985)
  - Typical applications to 8 in (52 cm) only (0.35 ft^2 (0.033 m^2))

Intermediate Break LOCA

- Not Analyzed, Historically Considered Non-Limiting

Status of ASTRUM Methodology Applications Internationally

- Significant international experience:
  - More than 75% PWR fleet in USA
  - Tihange 3 (Belgium)
  - Ringhals 3 (Sweden)
  - EDF CPY Class PWR (France) (Under review)
  - Angra 1 (Brazil)
  - Almaraz (Spain)
  - Maanshan (Taiwan)
  - AP1000 (China)

Robust Licensing Basis

Next Generation W LOCA Evaluation Model: Full Spectrum LOCA Methodology

- Development of a Realistic LOCA Methodology that Addresses All LOCA Break Sizes for W/CE Fleet in a Single Analysis
  - Development and V&V of WCOBRA/TRAC-TF2
- Development roadmap consistent with RG 1.203: Evaluation Model Development and Assessment Process (EMDAP)
- Build on 30 years of lesson learned
- Submitted to USNRC for their review and approval in 2010

Eliminate Small and Intermediate Breaks
as a Design Constraint

Development of Best-Estimate Plus Uncertainty Evaluation Models

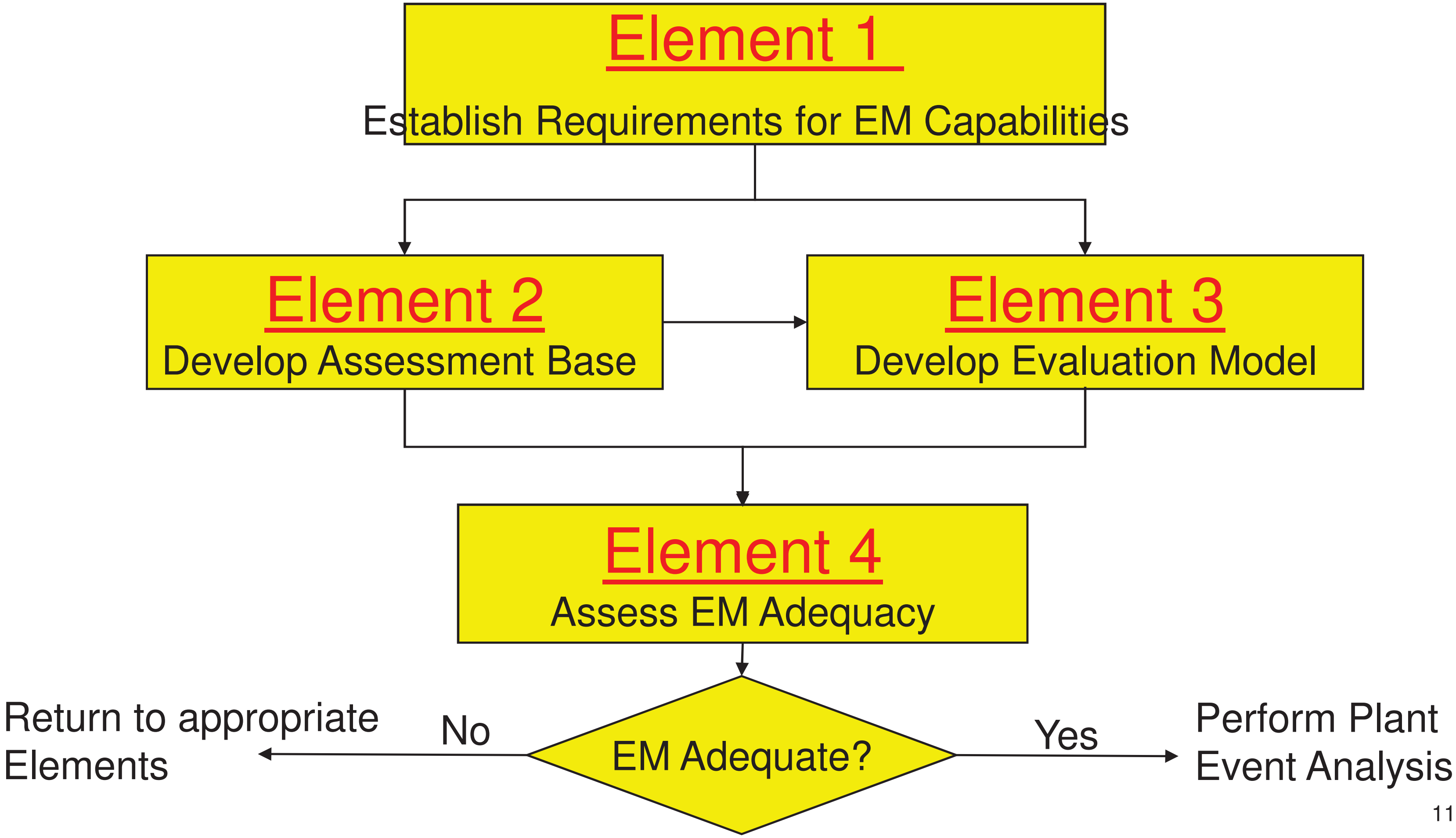
EMDAP Process (RG 1.203)

Framework for BEPU EM in Industry:

SRP NUREG-0800 and Regulatory Guide 1.203

- RG 1.203 was released in Dec. 2005.
- Expand on the same principles that were applied in CSAU (RG 1.157).
- Describes the “Evaluation Model Development and Assessment Process (EMDAP)”
- The process is applicable to any transient and accident analysis method (not limited to LOCA)

Regulatory Guide 1.203 - EMDAP



Element 1

Establish Requirements for EM Capabilities

Element 2

Develop Assessment Base

Element 3

Develop Evaluation Model

Element 4

Assess EM Adequacy

EM Adequate?

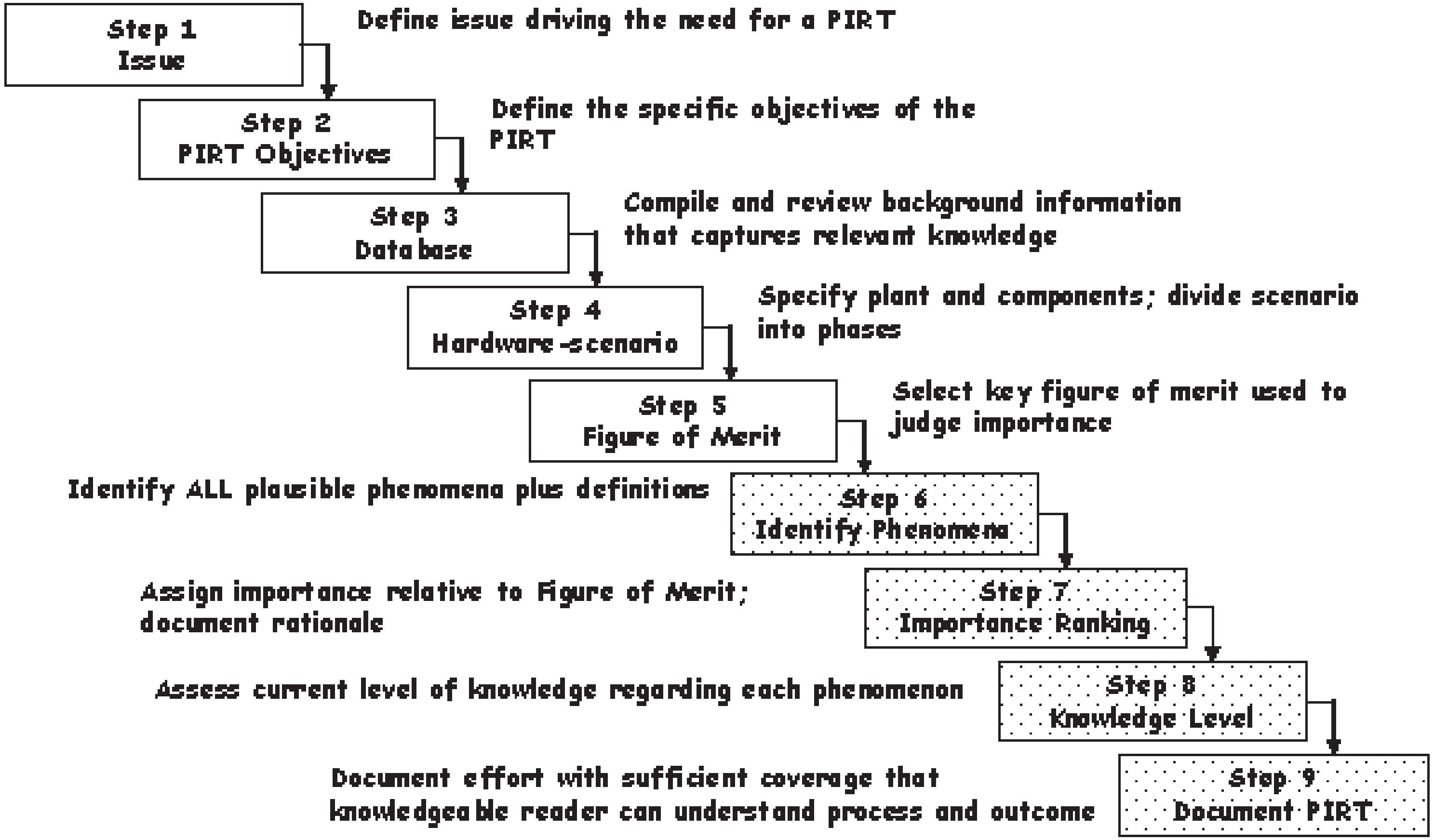
No

Return to appropriate
Elements

Yes

Perform Plant
Event Analysis

Starting Point in EMDAP:
Phenomena Identification and Ranking Table (PIRT)



Step 1
Issue

Define issue driving the need for a PIRT

Step 2
PIRT Objectives

Define the specific objectives of the PIRT

Step 3
Data base

Compile and review background information that captures relevant knowledge

Step 4
Hardware-scenario

Specify plant and components; divide scenario into phases

Step 5
Figure of Merit

Select key figure of merit used to judge importance

Step 6
Identify Phenomena

Identify ALL plausible phenomena plus definitions

Step 7
Importance Ranking

Assign importance relative to Figure of Merit; document rationale

Step 8
Knowledge Level

Assess current level of knowledge regarding each phenomenon

Step 9
Document PIRT

Document effort with sufficient coverage that knowledgeable reader can understand process and outcome

Boyack – Wilson, BE2004

FULL SPECTRUM LOCA

Westinghouse

From PIRT to EM Functional Requirements:
Code Evaluation process

- For each **medium** and **high** ranked phenomena, following questions are asked to identify and prioritize the important models for implementation consideration and/or assessment:
  1. Is there a physical model (or set of models) capable of describing a particular phenomenon in the EM?
  2. If **yes**, is the model validated to work in the desired range of conditions and geometries?
  3. Are there validation test data available (SETs and/or IETs) for validation of the model(s) in the expanded range?
  4. Do we need more test (SETs or IETs) to complete the assessment?

EMDAP (CSAU) Topics of Discussion Issue 1 – V&V Process

- Topic of debate in the industry for the V&V process is:
  - How to resolve apparent incongruence between the ‘attempt’ of modelling thermal-hydraulic processed at a very fine scale (with semi-mechanistic models) while both PIRT, SET and IET are only able to characterize an aggregate of such sub-processes.
  - T/H Code models are still empirical and based on a collection of correlations (closure relationships) which essentially are data fit and often not solving the fundamental physics underneath.
  - The data from the experiments is not detailed enough to characterize all those sub-process at the same level of the current computer code describes them.

EMDAP (CSAU) Topics of Discussion Issue 1 – Best Practice

- Indentify macroscopic controlling parameters whose bias and uncertainty is shown to aggregate the uncertainty associated with various individual sub-process modelled by the computer code.
- Demonstrate that the code is shown to capture bias and uncertainty by comparing the code prediction of rather ‘integral’ but ‘prototypical’ SETs often together with full scale data
- Perform detailed **compensating error analysis** to demonstrate that either the code is capable of predicting the right results for the right reasons, or
- if distortions are found, demonstrate that they are in the conservative direction to help the safety case for the plant under study.

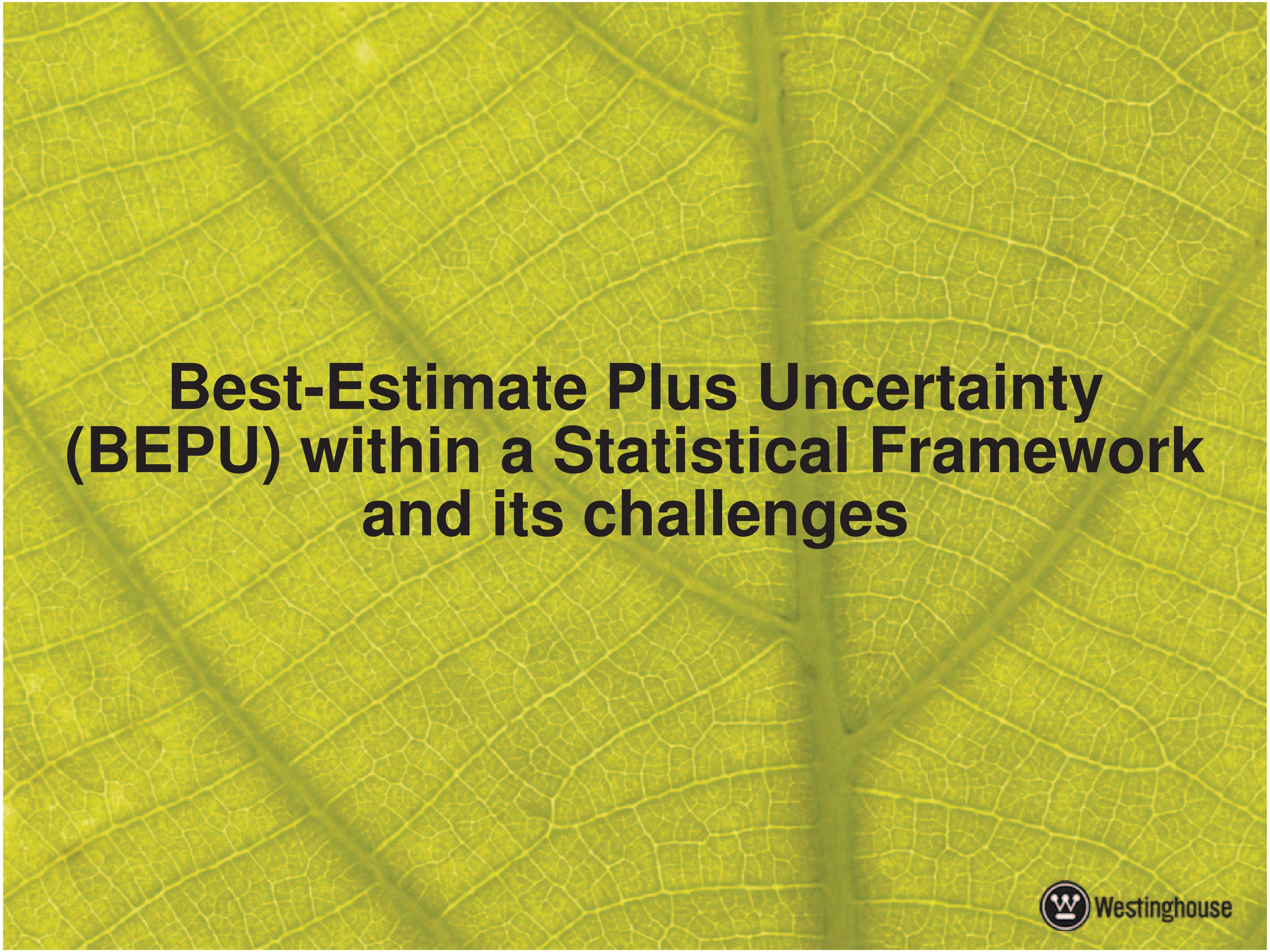
EMDAP (CSAU) Topics of Discussion Issue 2 – User Effect

- Since the introduction of BEPU methods, the influence of the user on the results has been recognized as another potential source of uncertainty.

- *Reference: “Aksan S. N., D'Auria F., Staedtke H, “User Effects on the Thermal-hydraulic Transient System Codes Calculations” J. Nuclear Engineering & Design, Vol 145, Nos. 1&2, 1993”*

EMDAP (CSAU) Topics of Discussion Issue 2 – Best Practice

- Eliminate such variability via several engineering safeguards or procedures that need to be integral part of the methodology.
  - Reinforce rigorous consistency between the noding utilized to model prototypical SET and IET and PWR.
  - Perform plant-specific inputs and set up of initial and boundary condition following a very prescriptive standard guidance (procedures).
  - Establish steady-state criteria to minimize variability of initial conditions
  - Limit plant-to-plant variations as much as practical and institute frequent engineering and peer reviews to assure adherence to guidance principles when differences arise.

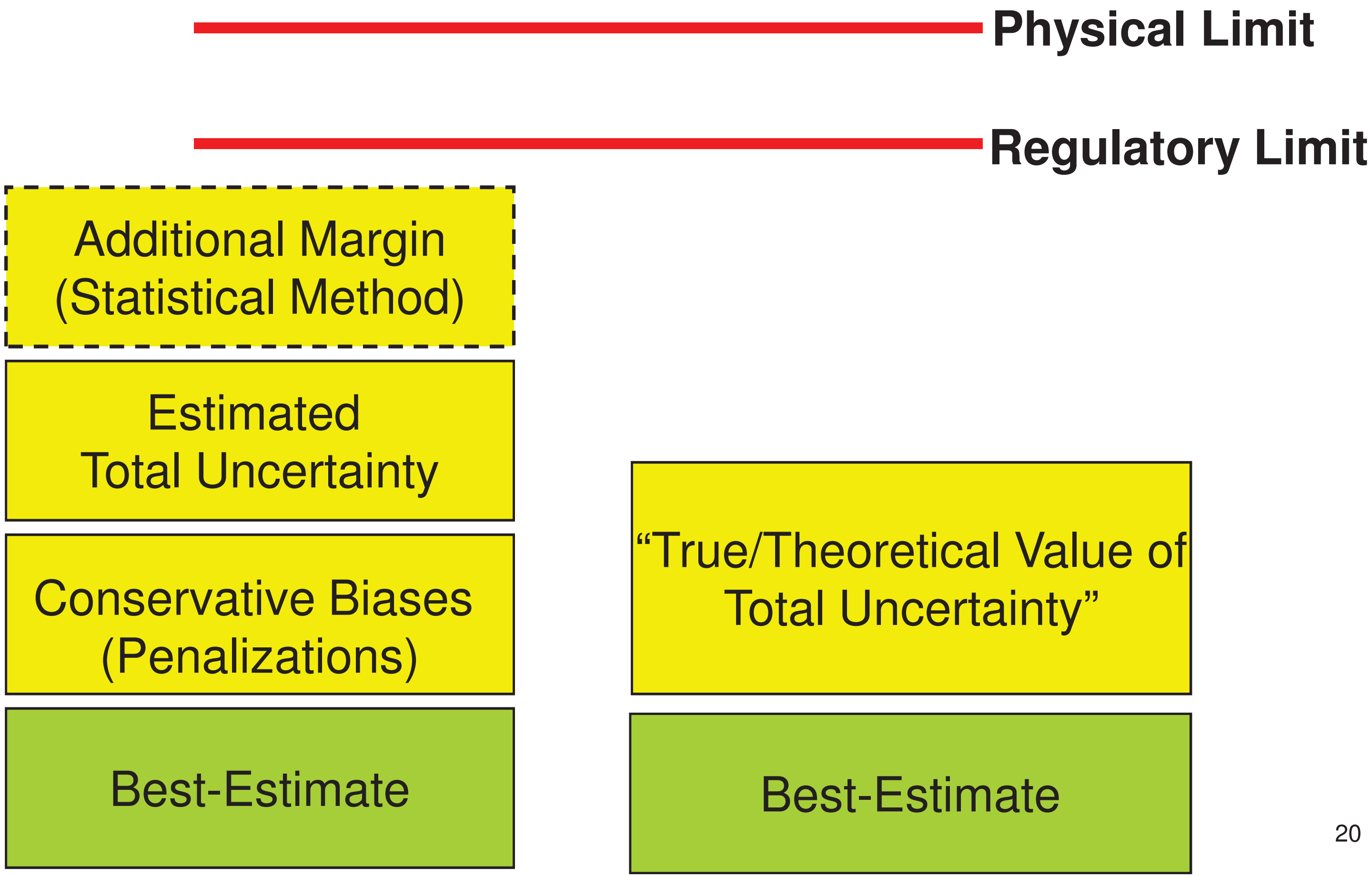


Best-Estimate Plus Uncertainty (BEPU) within a Statistical Framework and its challenges

10 CFR 50.46 Acceptance Criteria

- Peak clad temperature (PCT) < 2200 F
- Local maximum oxidation (LMO) < 17%
- Core-wide oxidation (CWO) < 1%
- The core should maintain a coolable geometry
- Long term cooling should be demonstrated

10 CFR 50.46 Criteria Compliance



Statistical Methods and 10 CFR 50.46 Compliance (95/95 criterion for LOCA Analyses)

- 10 CFR 50.46 requires that *"[…] uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section,* ***there is a high level of probability*** *that the criteria would not be exceeded."*
  - What is a "high level of probability"?
  - Is a 95/95 criterion (95th percentile at 95% confidence) adequate for LOCA analyses? And how is this adequacy justified?

Statistical Methods and 10 CFR 50.46 Compliance (95/95 criterion for LOCA Analyses)

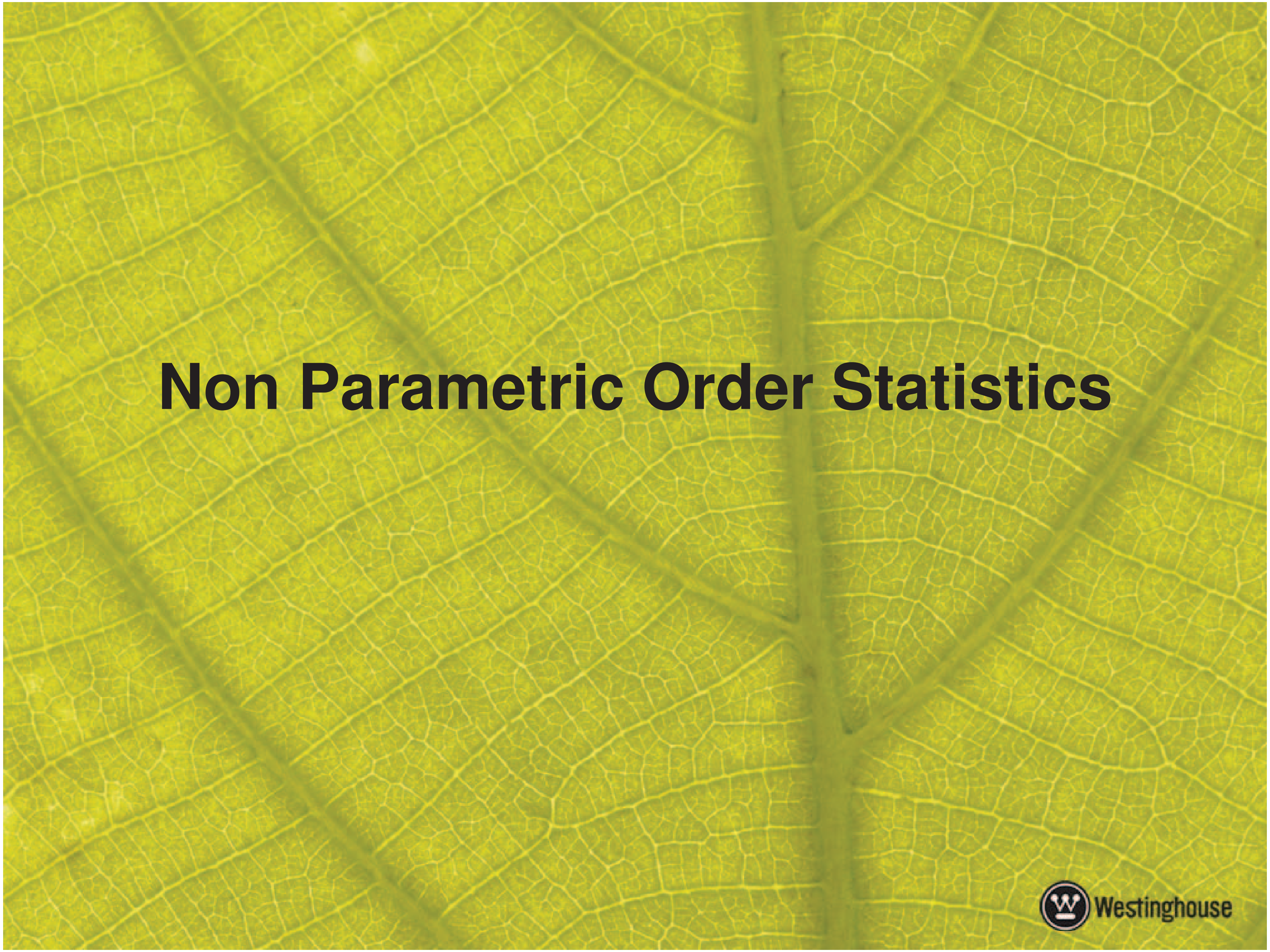
- Section 4 of Regulatory Guide 1.157 (Best-Estimate Calculations of Emergency Core Cooling System Performance) provides the NRC position on the “high probability”
  - *“A 95% probability is considered acceptable to the NRC staff […] to show that there is a high probability that the criteria [b.1 to b.3 of 10CFR50.46] will not be exceeded”*
- Regulatory Guide 1.157 simply introduces the concept of confidence level without elaborating further on what level is adequate.

Statistical Methods and 10 CFR 50.46 Compliance (95/95 criterion for LOCA Analyses)

- *"[…] the staff determined that a 95th percentile probability level based on best approximations of the constituent parameter distributions <u>and the statistical approach used in the methodology</u> is appropriately high"*
- *" Because this application only applies to LBLOCA design basis analyses (which assume a single failure), a higher probability […] is not needed to assure a safe design."*
- *CQD SER – "[…] further conservatism in the code prediction does not contribute to an increase in public health and safety"*

“Conservatisms” in Best Estimate Methods

- The single failure assumption is not the only conservative bias/assumption included in the Westinghouse methodology
  - Conservatisms (bias) in code predictions of ECC bypass, entrainment and steam binding
  - Methods used to establish blowdown cooling and reflood heat transfer multipliers and limits
  - Conservative treatment of peaking factors (F_Q assumes plant is always in a transient, nominal $F\Delta H$ is set to maximum allowed)
  - Worst assumption for off-site power availability
  - Minimum containment pressure
  - Others (conservative fuel temperatures, minimum safety injection,…)



Non Parametric Order Statistics

Westinghouse

Westinghouse BEPU EM (ASTRUM) is based on Guba and Makai (2003)

- Guba and Makai (2003) generalized the Wilks's Method for cases when p>1 (multivariate). The size of the sample is determined by the following equation:

$$\beta = \sum_{j=0}^{N-p} \frac{N!}{(N-j)!\,j!} \gamma^j (1-\gamma)^{N-j}$$

- Where:
  - β = confidence level
  - N = sample size (number of runs)
  - p = number of output variables
  - γ = tolerance interval

Westinghouse BEPU EM (ASTRUM) is based on Guba and Makai (2003)

- Guba results reduce to Wilks's when p=1(one variate).
- In particular for β = 0.95 and γ = 0.95, we obtain:
  - p = 1 (i.e. PCT) → **N = 59**
  - p = 3 (i.e. PCT, LMO, CWO) → **N = 124**

PCT = Peak Clad Temperature
LMO = Local Maximum Oxidation
CWO = Core-Wide Oxidation

Debate on Use of Order Statistics in Industry/Academia

- A significant debate has taken place in the technical community regarding the practical implementation of Order Statistics.
  - The focus of this debate has been on the actual number of runs required to satisfy the LOCA licensing criteria.
  - In this framework, Westinghouse approach has been considered (overly) conservative.
  - Various authors have suggested that a reduced number of runs (compared to 124) may be sufficient to develop a 'statistical statement' which satisfies the LOCA licensing criteria.

Debate on Use of Order Statistics in Industry/Academia

- Regardless the different embodiments that can be found within the industry, the reliance on the minimum sample size and use the extreme case (rank k=1) as upper tolerance limit presents some challenges:
  - The extreme case is subject to **large variability**.
  - The estimator is **affected by the seed** selected during the Monte Carlo procedure.
  - When we aim to a 95/95 upper tolerance limit, we are willing to accept that we can have **5% probability of being in error** in our inference of bounding the 95th percentile. This presents some challenges to some regulators.

Debate on Use of Order Statistics in Industry/Academia

- The problem is somewhat alleviated within the Westinghouse embodiment (ASTRUM) because the procedure aims on a 95/95 for three variables (max of 124 instead of max of 59).
  - In that case, on a single output the probability of underestimation is reduced to 0.2%.
- It is often not practical to perform parameter sensitivities, or regressions analyses with small sample sizes.
  - The sample size is often insufficient relative to the magnitude of the spread of the data in the sample.
- **Many of these issues are alleviated by simply increasing the sample size.**

Conclusions

- Licensing-grade BEPU methods are now widespread in the nuclear industry, particularly for LOCA safety analysis.
- The preferred framework of these methods has been the CSAU, more recently reformulated in the Evaluation Model Development and Assessment Process (EMDAP) (RG 1.203).
- Many lesson learned have been accumulated over the past three decades and captured within the new generation of EM (e.g. in Westinghouse Full Spectrum LOCA Methodology)

Moving Forward

- Many challenges still lie ahead and there is significant room for further improvement.
  - These methods still retain significant conservatism in their model in how they are applied for nuclear power plant safety analysis.
  - Applications of approved methods are valid only within the constraints of the class of plants and scenarios for which they are certified.
  - As new designs and new scenarios are identified, this requires continuous development and assessment.
- The area of uncertainty, particularly the use of non-parametric order statics has been a success story within the industry; however both applicants and regulators are aware of its shortcomings.

Questions

?

RELAP5/MOD3.2 Sensitivity Analysis Using OECD/NEA ROSA-2 Project 17% Cold Leg Intermediate-break LOCA Test Data

November 17, 2011

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

Nuclear Safety Research Center,
Japan Atomic Energy Agency (JAEA)

T. TAKEDA, T. WATANABE, Y. MARUYAMA, H. NAKAMURA

Contents

- Background & Objectives
- LSTF Facility & OECD/NEA ROSA-2 Project
- Test & RELAP5 Post-test Analysis Conditions
- Comparison of Test & Post-test Analysis Results
- Selection of Key Phenomena & Important Parameters
- Selection of Ranges of Important Parameters
- RELAP5 Sensitivity Analysis Results
- Conclusions

Background

- Frequency of DEGB of PWR main cooling pipe *leading to LBLOCA*; quite low
- Risk informed regulation-relevant safety analysis
- ✓ Consideration of IBLOCA; relatively more important than ever
- Realistic evaluation of safety margin in IBLOCA
  - ✓ Conduct OECD/NEA ROSA-2 Project 17% cold leg IBLOCA test using LSTF facility to simulate DEGB of ECCS nozzle

Objectives

- Validate predictability of RELAP5/MOD3.2.1.2 code, to be used as platform for **BEPU study**, to T/H phenomena by post-test analysis of LSTF test
- Select key phenomena & related important parameters based on LSTF test data analysis & RELAP5 post-test analysis results
- Investigate influences of important parameters relevant to key phenomena onto cladding surface temperature by RELAP5 sensitivity analysis of LSTF test

DEGB: Double-ended Guillotine Break, IBLOCA: Intermediate-break LOCA

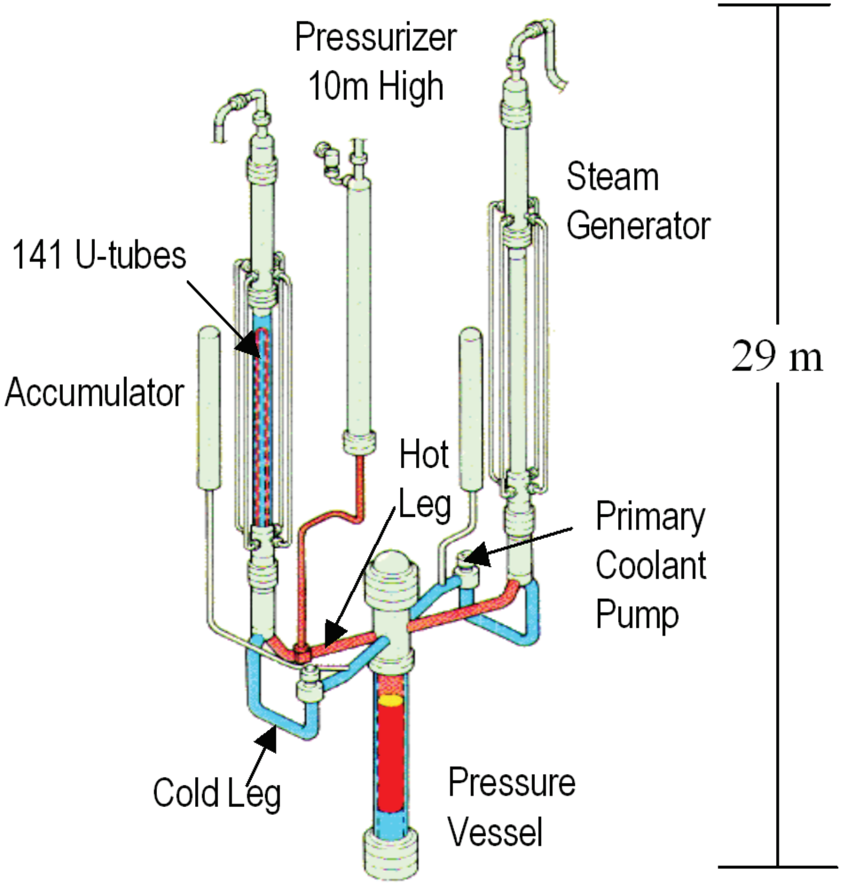
Large Scale Test Facility (LSTF) at JAEA

Simulate W-type four-loop 3423 MWt PWR (Tsuruga Unit-2 of JAPC)

- ✓ Full-height and 1/48 in volume
- ✓ Two loops each including SG, primary coolant pump, hot and cold legs
- ✓ 141 full-size U-tubes in each SG
- ✓ 10 MW core power (14% of 1/48-scaled PWR rated value)

OECD/NEA ROSA-2 Project

- ✓ **Resolve issues in T/H analyses relevant to LWR safety by using LSTF**
- ✓ More than 18 organizations from 14 NEA member countries (from 2009 to date)
- ✓ Provide datailed T/H data with complex phenomena suitable for validation of computer codes and models



Schematic of LSTF

LSTF Test Conditions

Break Conditions

- Break size of 17% at cold leg in broken loop without PZR
  - ✓ 41 mm i.d. nozzle upwardly mounted flush with cold leg inner surface

ECCS Conditions

- Actuations of HPI, ACC & LPI systems in intact loop only to simulate ECCS line break
- Single-failure of diesel generators related to HPI & LPI flow rates

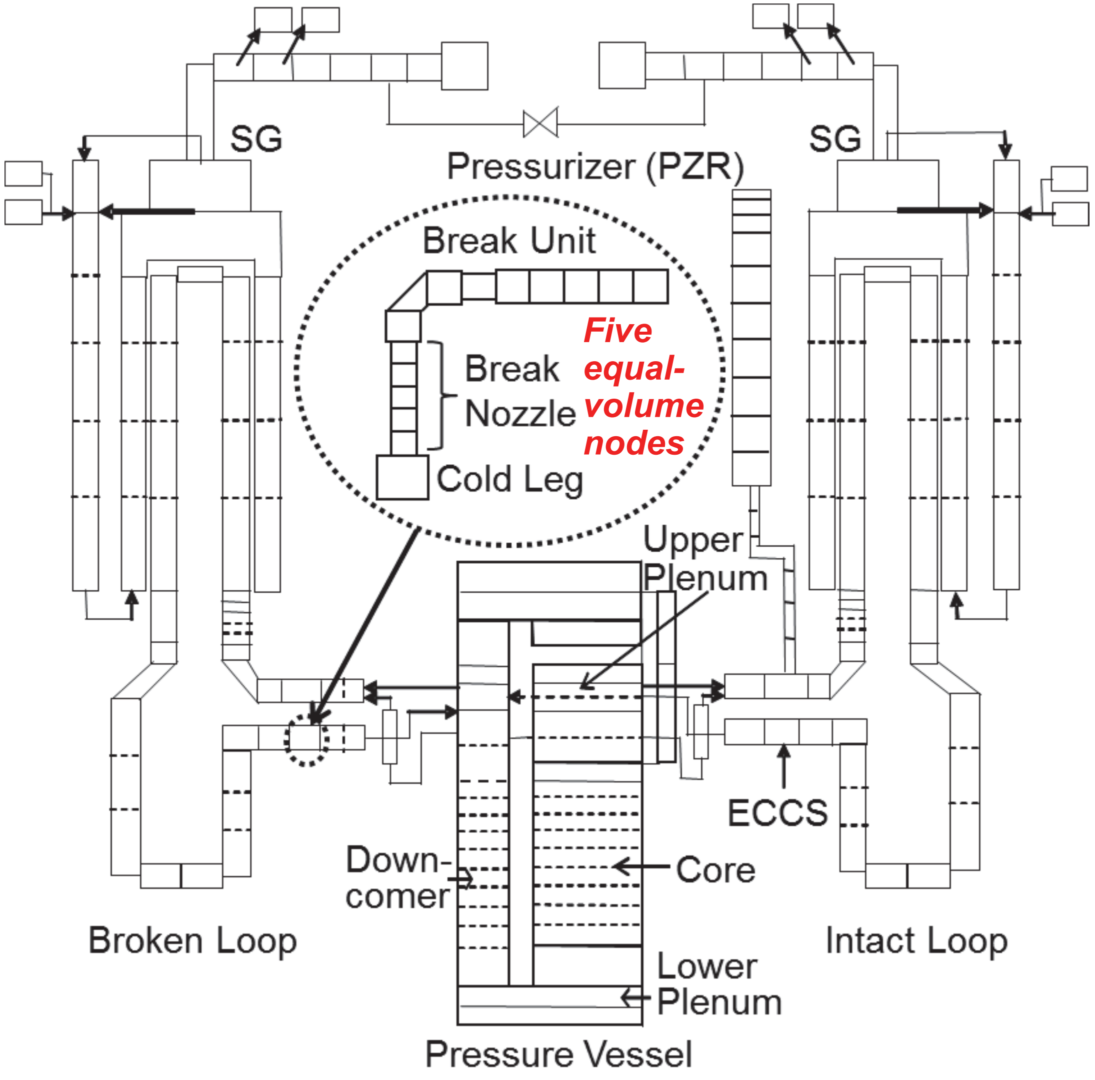
Assumptions

- Total failure of auxiliary feedwater
- Loss of off-site power concurrent with scram
- Following threshold temperature of maximum fuel rod surface temperature for LSTF core protection & power controlling sytsem

958K=70% of pre-determined value, 961K=35%, 966K=13%, 977K=5%, 1003K=0%

HPI: High-pressure injection, ACC: Accumulator injection, LPI: Low-pressure injection

RELAP5 Code Post-test Analysis Conditions (as 'Base Case')



Noding of LSTF for RELAP5 Analysis

- Employ break flow model of RELAP5/MOD3.2.1.2 code for long break nozzle, with discharge coefficient (Cd) of 1.0
- Reduce gas-liquid inter-phase drag to 1/10 in core, referring to Kumamaru's study*
- Apply Wallis CCFL correlation to core exit, inlet plena & U-tubes of SGs, based on test results

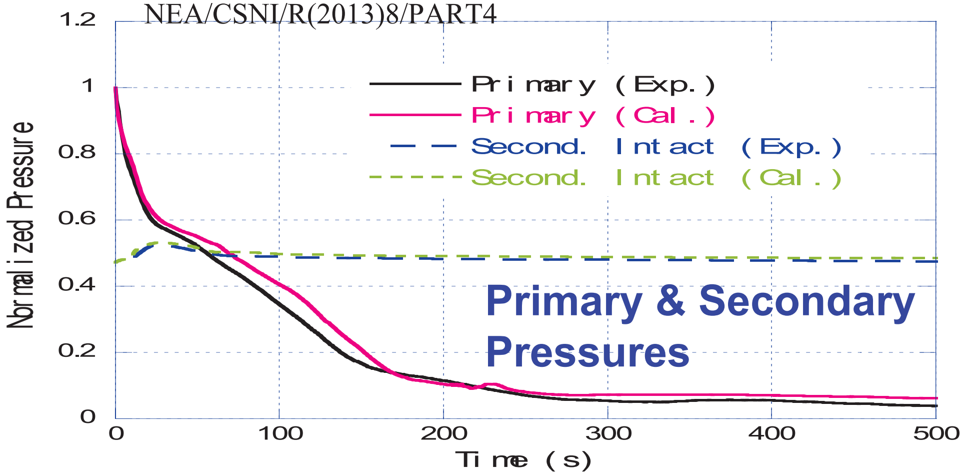
$$j_g{}^{*1/2} + m j_\ell{}^{*1/2} = C$$

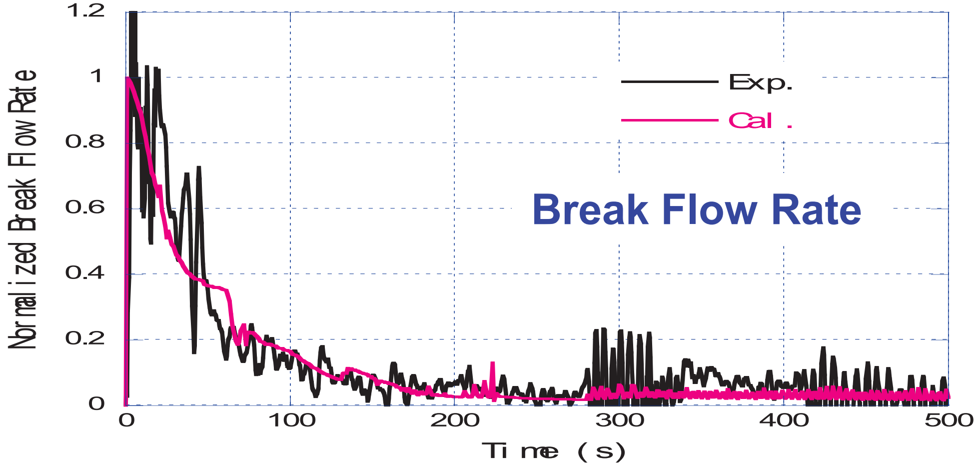
j : Non-dimensional volumetric flux,*
*m, C: Constants**; m=1,C=0.75*

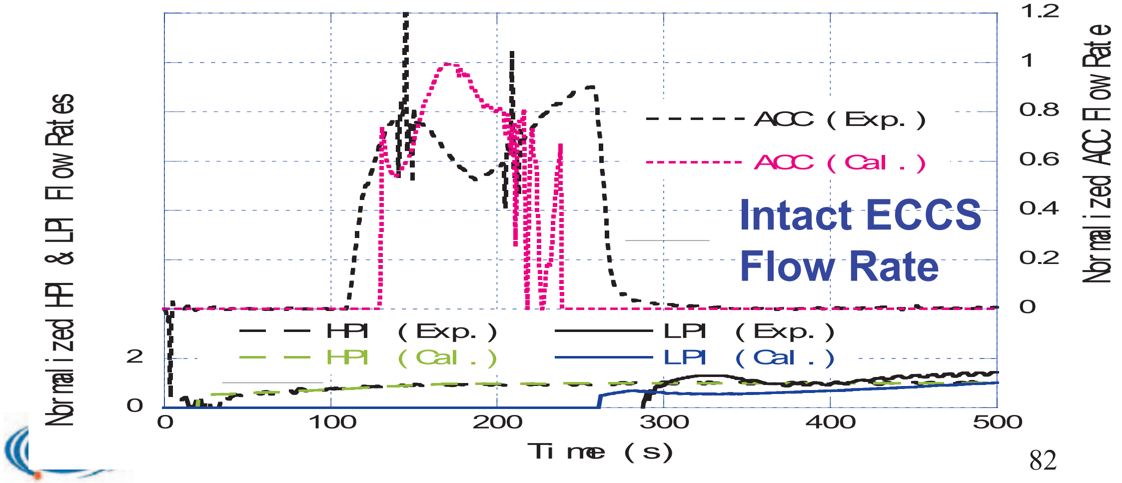
- No core power control depending on PCT *(higher than 958 K)*

* Kumamaru, H., *et al.*, Nucl. Technol. Vol.126 (1999).
** Yonomoto, T., *et al.*, Proc. of ANS Int. Mtg. (1991).

Test & RELAP5 Post-test Analysis Results (1/2)

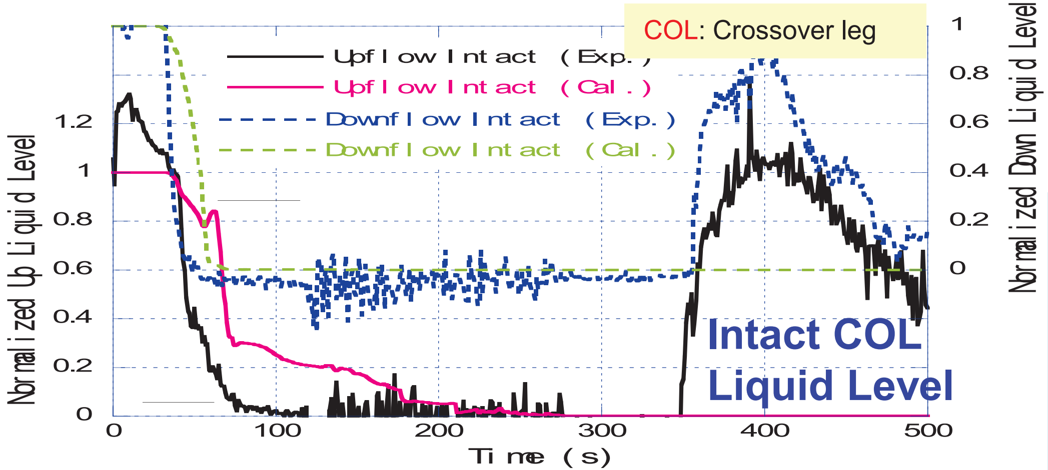


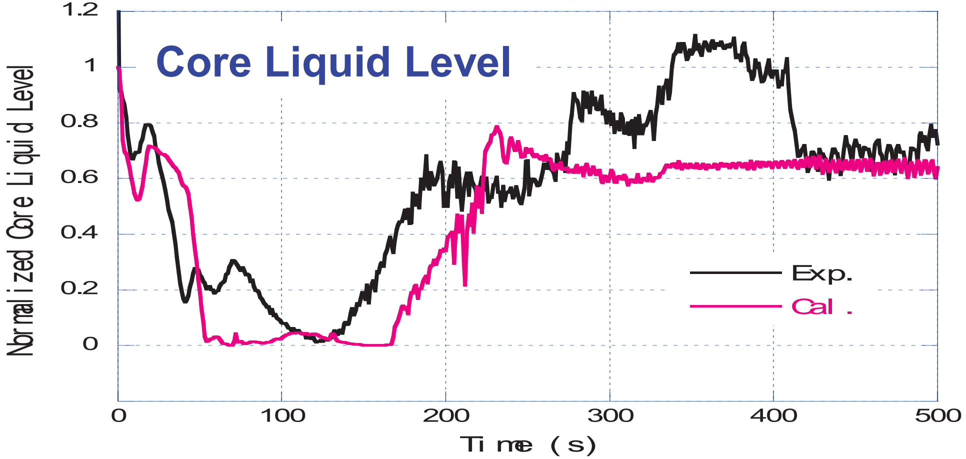


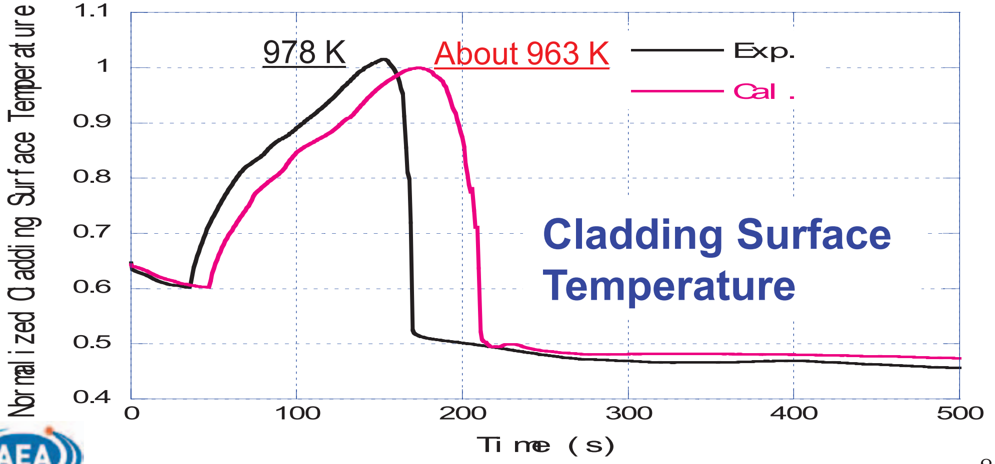


- ➢ Fast primary depressurization due to rather large size of break
- ➢ HPI system was started almost simultaneously with core dryout, but was ineffective on core cooling due to far smaller injection flow rate than break flow rate.
- ➢ Overprediction of primary pressure due to smaller steam discharge through break resulting in later ACC coolant injection than in test, though reasonably-well predictions of break flow rate and HPI flow rate

Test & RELAP5 Post-test Analysis Results (2/2)







- Core dryout took place due to rapid drop in core liquid level before LSC.
- Liquid was accumulated in upper plenum, SG U-tube upflow-side & SG inlet plenum before LSC due to CCFL, causing further drop in core liquid level.
- Large temperature excursion appeared in core even after reflooding due to ACC injection, causing PCT was 978 K at Pos. 6 with core power down to 5%.
- Later LSC than in test
- After around 50 s, underprediction of core liquid level due to overprediction of upper plenum liquid level under effects of CCFL at core exit
- Underprediction of cladding surface temperature due to later core uncovery than in test, causing PCT was about 963 K at Pos. 5 (=core center)

RELAP5 Sensitivity Analysis of LSTF Test

'Base Case' for Sensitivity Analysis

- Post-test analysis conditions were considered as 'Base Case', for sensitivity analysis to study causes of uncertainty in best estimate methodology.

Objective of Sensitivity Analysis

- Investigate influences of important parameters relevant to key parameters onto cladding surface temperature

Selection of Key Phenomena & Important Parameters

- Affect core liquid level behavior and thus cladding surface temperature
- Based on LSTF test data analysis and RELAP5 post-test analysis results
- Refer to high-ranked PWR LOCA phenomena*

* Boyack and Ward, Proc. Int. Mtg. BE-2000 (2000).

Selection of Key Phenomena & Important Parameters

Component	Key Phenomenon	Rank*	Important Parameter	Classification
Break	Critical flow	H	Break discharge coefficient (Cd)	Physical model
Fuel rod	Decay heat	H	Core decay power	Boundary condition
	Stored heat	M	Thermal conductivity of fuel rod	Material property
			Heat capacity of fuel rod	
Core	Mixture level	H	Gas-liquid inter-phase drag in core	Physical model
Upper plenum	Liquid accumulation due to CCFL**	H	Constant C of Wallis CCFL correlation at core exit	
SG U-tubes		H	Constant C of Wallis CCFL correlation at SG U-tubes	
SG inlet plena		M	Constant C of Wallis CCFL correlation at SG inlet plena	
Cold leg / HPI or ACC system	Steam condensation on HPI or ACC coolant	M	Condensation heat transfer with Shah's model	
			Injection pressure of HPI system	Boundary condition
			Injection temperature of HPI system	
			Injection pressure of ACC system	
			Injection temperature of ACC system	

*H; High-ranked phenomenon, M; Medium-ranked phenomenon **Phenomenon peculiar to cold leg IBLOCA

- *High-ranked (H) phenomenon*; great influences onto core liquid level behavior and thus cladding surface temperature
- *Medium-ranked (M) phenomenon*; somewhat or little influences onto core liquid level behavior and thus cladding surface temperature
- Need sensitivity analysis for *both high- and medium-ranked phenomena*

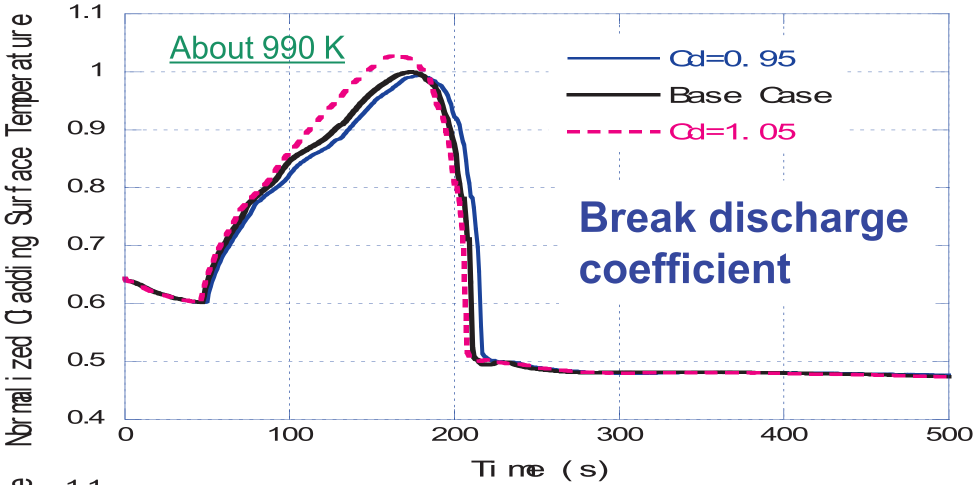
Selection of Ranges of Important Parameters

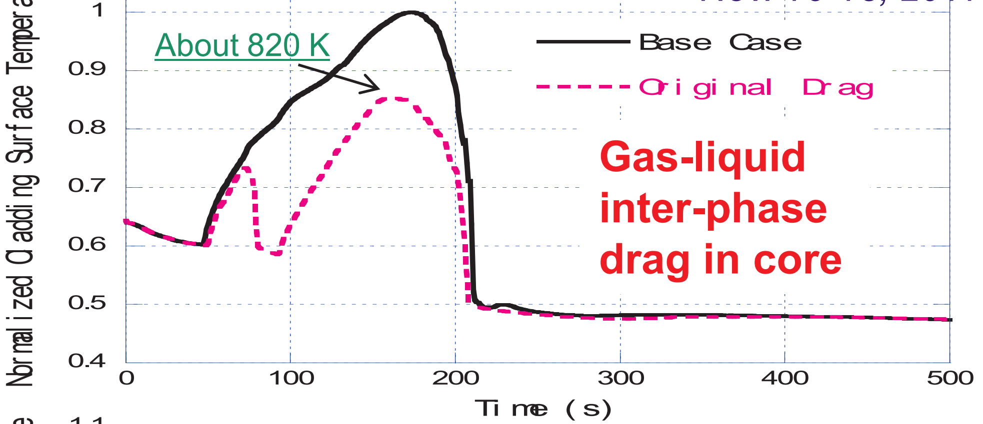
Important Parameter	Range	**Rank** for Phenomenon	**Band**
Break discharge coefficient (Cd)	1.0±0.05	H	Small
Core decay power	Specified±0.07 MW*	H	
Thermal conductivity of fuel rod	Specified±10%	M	Refer to database
Heat capacity of fuel rod	Specified±1%	M	Refer to database
Gas-liquid inter-phase drag in core	1/10, 1	H	Base Case & origin.
Constant C of Wallis CCFL correlation at core exit	0.75±0.1	H	Small
Constant C of Wallis CCFL correlation at SG U-tubes	0.75±0.1	H	
Constant C of Wallis CCFL correlation at SG inlet plena	0.75±0.1	M	
Condensation heat transfer with Shah's model	1±0.5	M	Large
Injection pressure of HPI system	12.27±0.11 MPa*	M	
Injection temperature of HPI system	310±2.4 K*	M	
Injection pressure of ACC system	4.51±0.05 MPa*	M	
Injection temperature of ACC system	320±2.3 K*	M	

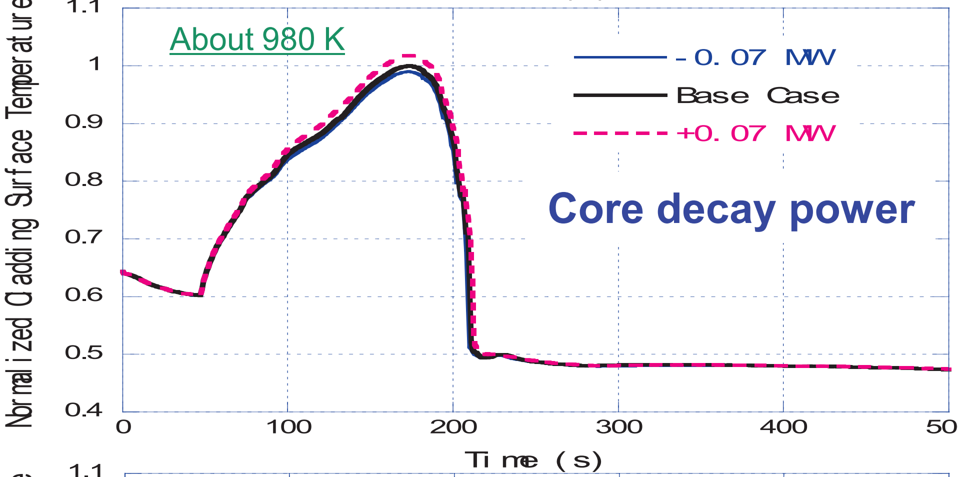
*Measurement uncertainty

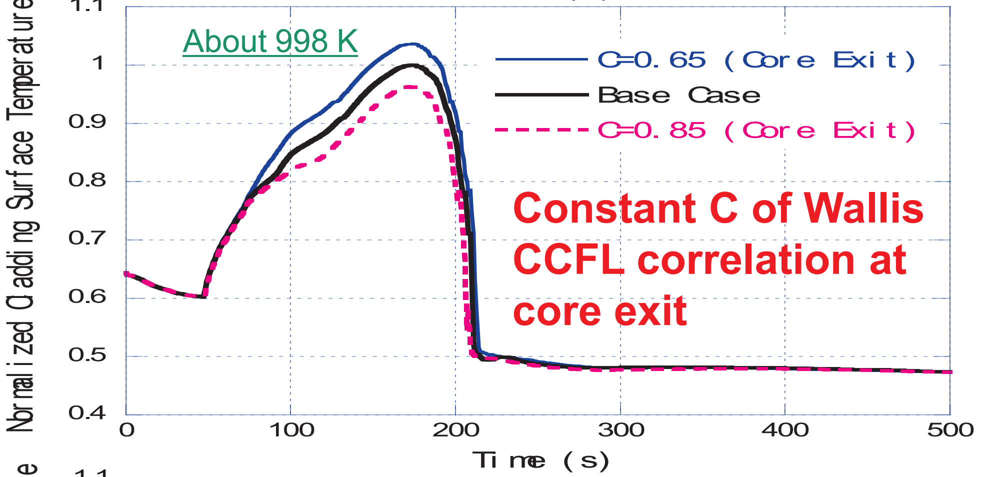
- Refer to values of 'Base Case'
- *Parameter related to high-ranked (H) phenomenon*; clarify influences of small difference in parameter onto cladding surface temperature
- *Parameter related to medium-ranked (M) phenomenon*; clarify influences of large difference in parameter onto cladding surface temperature

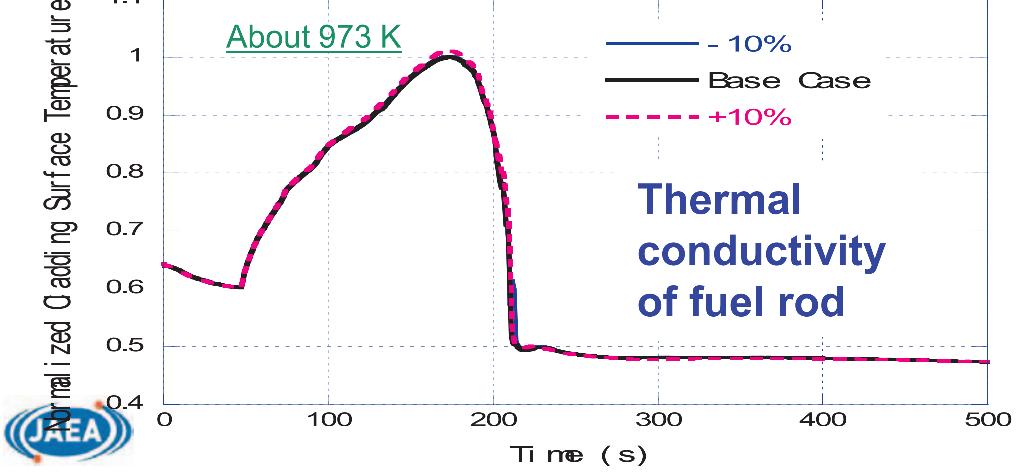
Sensitivity Analysis Results

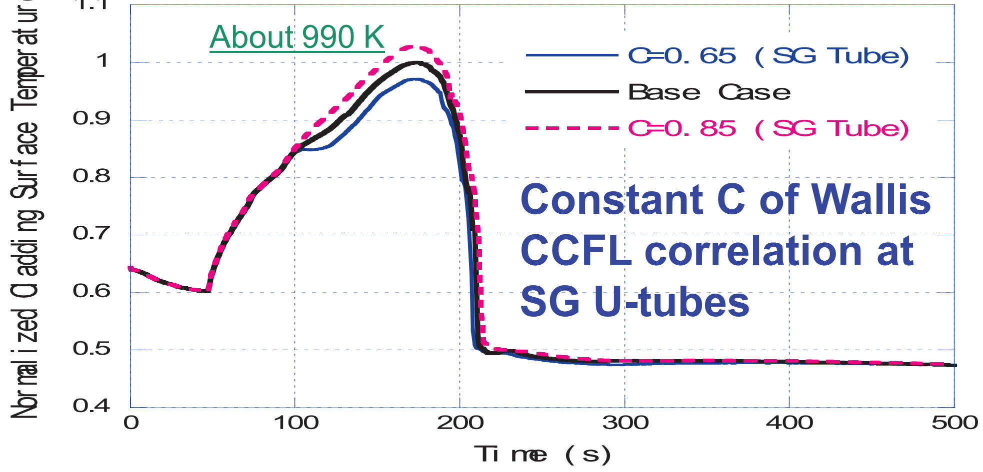












Summary of Sensitivity Analysis Results

Findings from Sensitivity Analysis

- Both constant C of Wallis CCFL correlation at core exit & gas-liquid inter-phase drag in core were more sensitive to cladding surface temperature than other chosen parameters.
  - ✓ Both CCFL at core exit & reduction of gas-liquid inter-phase drag in core affected cladding surface temperature significantly.
- Following parameters had little influences onto cladding surface temperature: *heat capacity of fuel rod, condensation heat transfer with Shah's model, constant C of Wallis CCFL correlation at SG inlet plena, ECCS injection conditions*

  => *Reconsider related phenomenon as low-ranked (L) phenomenon*

Future Work

- To be investigated influences of <u>combination especially for parameters related to high-ranked phenomena</u> onto cladding surface temperature through further sensitivity analysis

Conclusions

LSTF 17% Cold Leg IBLOCA Test

- Core dryout took place due to rapid drop in core liquid level before LSC.
- Liquid was accumulated in upper plenum, SG U-tube upflow-side and SG inlet plenum before LSC due to CCFL, causing further decrease in core liquie level.
- Large temperature excursion appeared in core even after reflooding due to ACC coolant injection, causing PCT was 978 K at Pos. 6 with core power down to 5%.

RELAP5 Post-test Analysis of LSTF Test

- Cladding surface temperature was underpredicted due to later core uncovery.

RELAP5 Sensitivity Analysis of LSTF Test

- Post-test analysis conditions were considered as ‘Base Case’, for sensitivity analysis to study causes of uncertainty in best estimate methodology.
- Critical flow at break, decay heat of fuel rod, mixture level in core, liquid accumulation in upper plenum and SG U-tube upflow-side were identified as high-ranked phenomena, based on LSTF test data analysis & post-test analysis results.
- Both constant C of Wallis CCFL correlation at core exit and inter-phase drag in core, as parameters that need to consider for evaluation of safety margin, were more sensitive to cladding surface temperature than other chosen parameters.

Addendum

Future Strategy

BEPU Study Using LSTF 17% Cold Leg IBLOCA Test Data

- **Creation of PIRT**, *for phenomena affected core liquid level behavior and thus cladding surface temperature based on present study*
- **Selection of uncertainty parameters & ranges**, *based on present study*
- **Uncertainty analysis with RELAP5 code**
  - Order statistic method; the number of calculations based on Wilks' formula

    Ex. 124; third order for 95% probability with 95% confidence level
  - Estimation of peak cladding surface temperature

Application of BEPU Methodology to Reference PWR (Tsuruga Unit-2)

PIRT: Phenomena Identification and Ranking Table

TRACTEBEL Engineering
GDF SUEZ

Towards an Industrial Application of Statistical Uncertainty Analysis Methods to Multi-physical Modelling and Safety Analyses

OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations,
Barcelona, Spain, 16-18 November 2011

Jinzhao Zhang
Jacobo Segurado
Christophe Schneidesch

CHOOSE EXPERTS, FIND PARTNERS

OUTLINE

- TE mission and needs
- TE current safety analysis capability
- TE Best Estimate plus Statistical Uncertainty Analysis Method (BESUAM)
- Choice of the statistical uncertainty analysis methods and tool
- Preliminary applications and results
- Conclusions and Perspectives

INTRODUCTION
> TE MISSION AND NEEDS

- Tractebel Engineering (TE)

  Architect and Owner's Engineer for all 7 Belgian NPPS

- TE mission
  - To provide safe, competitive and optimized solutions to the Utility
  - To answer the specific concerns of the Belgian Safety Authorities

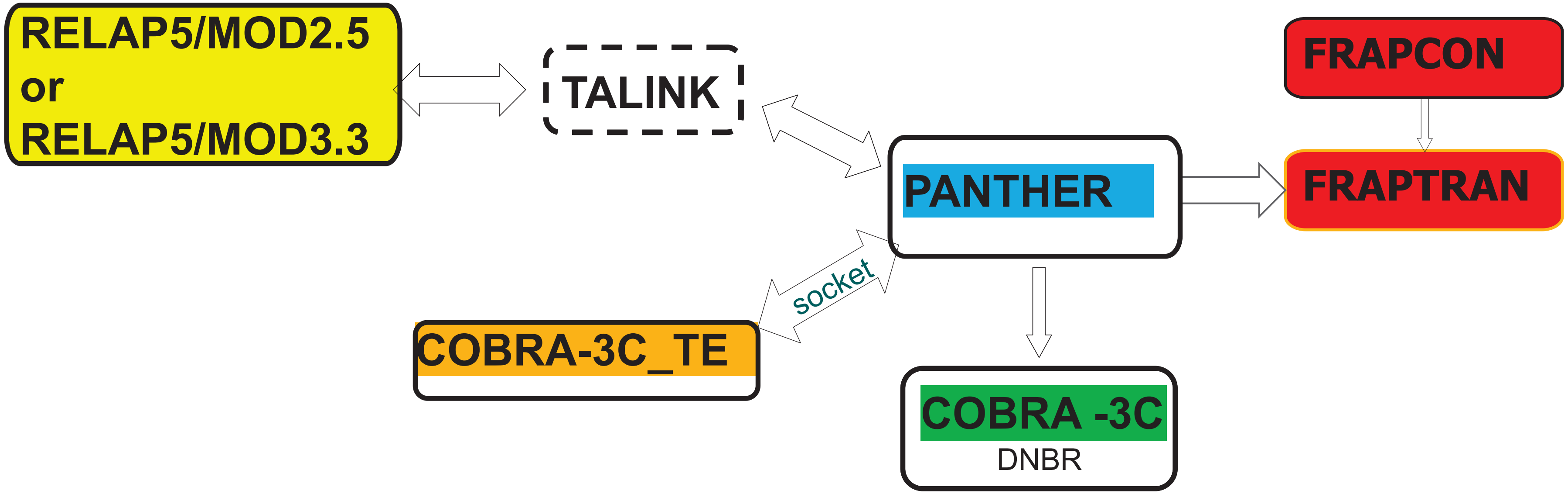
- Objective: To develop, license and apply BEPU methodologies for reload fuel safety evaluation and safety analysis

  → need quantification of the multi-physics code uncertainties (UQ)

  → need a simple, transparent, robust and flexible uncertainty analysis method (UAM)

CURRENT SAFETY ANALYSIS CAPABILITY

> MULTI-PHYSICS CODE PACKAGE



CURRENT SAFETY ANALYSIS CAPABILITY

> DETERMINISTIC BOUNDING APPROACH

- Use of « best-estimate » multi-physics code package
  - Review of known code deficiencies and uncertainties → corrected or bounded
- Determination of conservative assumptions (IC/BC)
  - Engineering judgement
  - Single-parametric sensitivity studies
- Deterministic combination of all uncertainties and conservative assumptions in a Licensing case
  - Applied to all non-LOCA accident analyses
  - Approved by the Belgian Safety Authorities

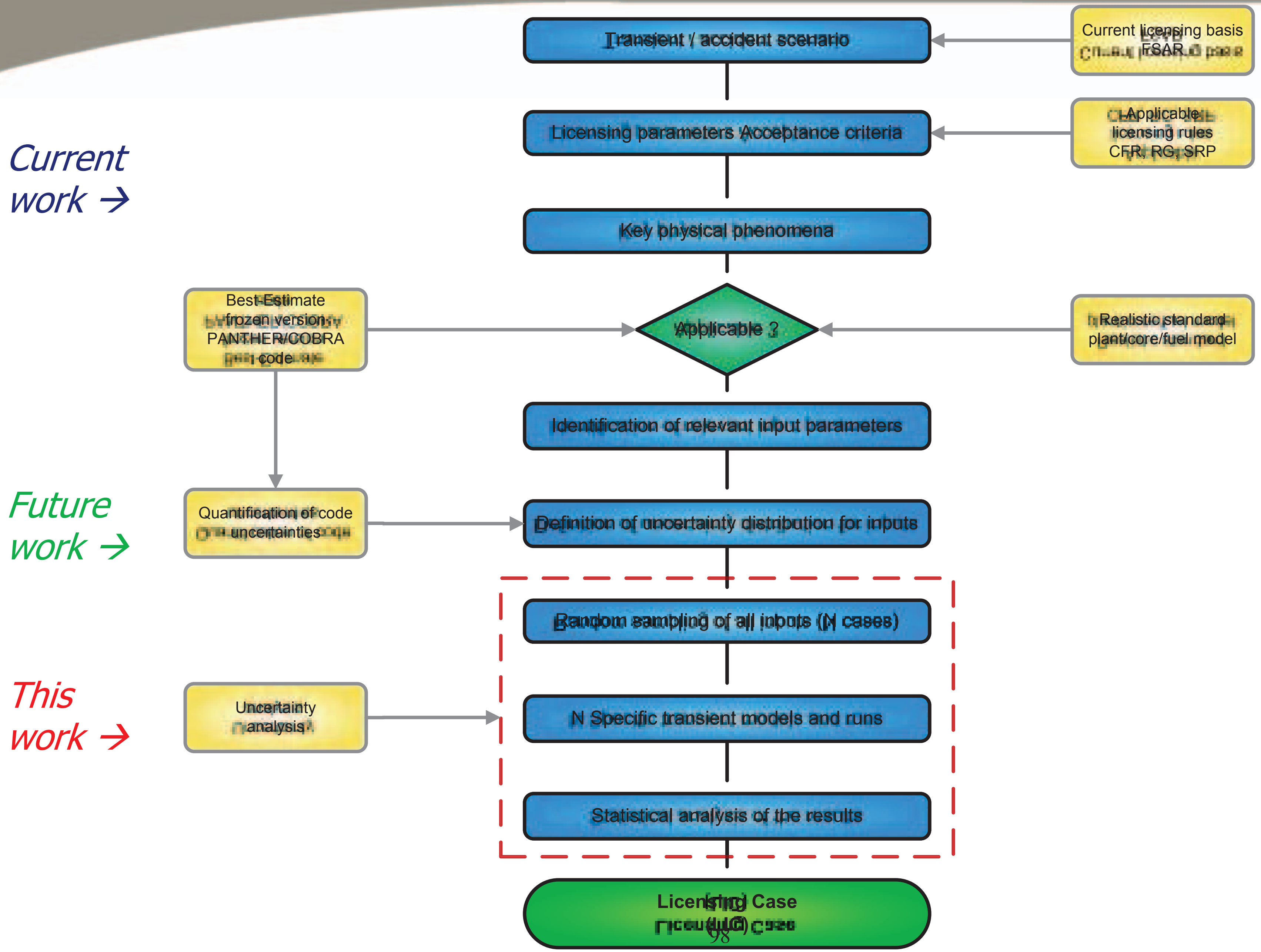
CURRENT SAFETY ANALYSIS CAPABILITY

> BENEFITS AND LIMITATIONS

- Consistent with the initial licensing basis
  - →easily acceptable by the Safety Authorities
- Based on engineering judgements rather than on rigorous uncertainty analysis
  - →needs a high level of expertise for methodology development
- Cost-effective for applications
  - → needs a large number of parametric studies for methodology development
- Same uncertainties and conservatism's
  - → may involve certain un-quantified conservatisms

PROPOSED BEST ESTIMATE PLUS STATISTICAL UNCERTAINTY ANALYSIS METHOD (BESUAM)

- Based on best estimate multi-physics code package
  - Extensive Verification & Validation (V&V) to quantify the code uncertainties (UQ) → *Future work*
- Use of statistical uncertainty analysis method
  - Statistical combination of uncertainties in plant conditions and code calculations
  - « Once-through » methodology; less assumptions, less expert judgement
- Objectives
  - to gain licensing margins with respect to the safety criteria
  - to keep in pace with the regulatory requirements (RG.1.205) and industry development trends



Transient / accident scenario

Current licensing basis
FSAR

Licensing parameters Acceptance criteria

Applicable
licensing rules
CFR, RG, SRP

Current work →

Key physical phenomena

Best-Estimate
frozen version
PANTHER/COBRA
code

Applicable ?

Realistic standard
plant/core/fuel model

Identification of relevant input parameters

Future work →

Quantification of code
uncertainties

Definition of uncertainty distribution for inputs

Random sampling of all inputs (N cases)

This work →

Uncertainty
analysis

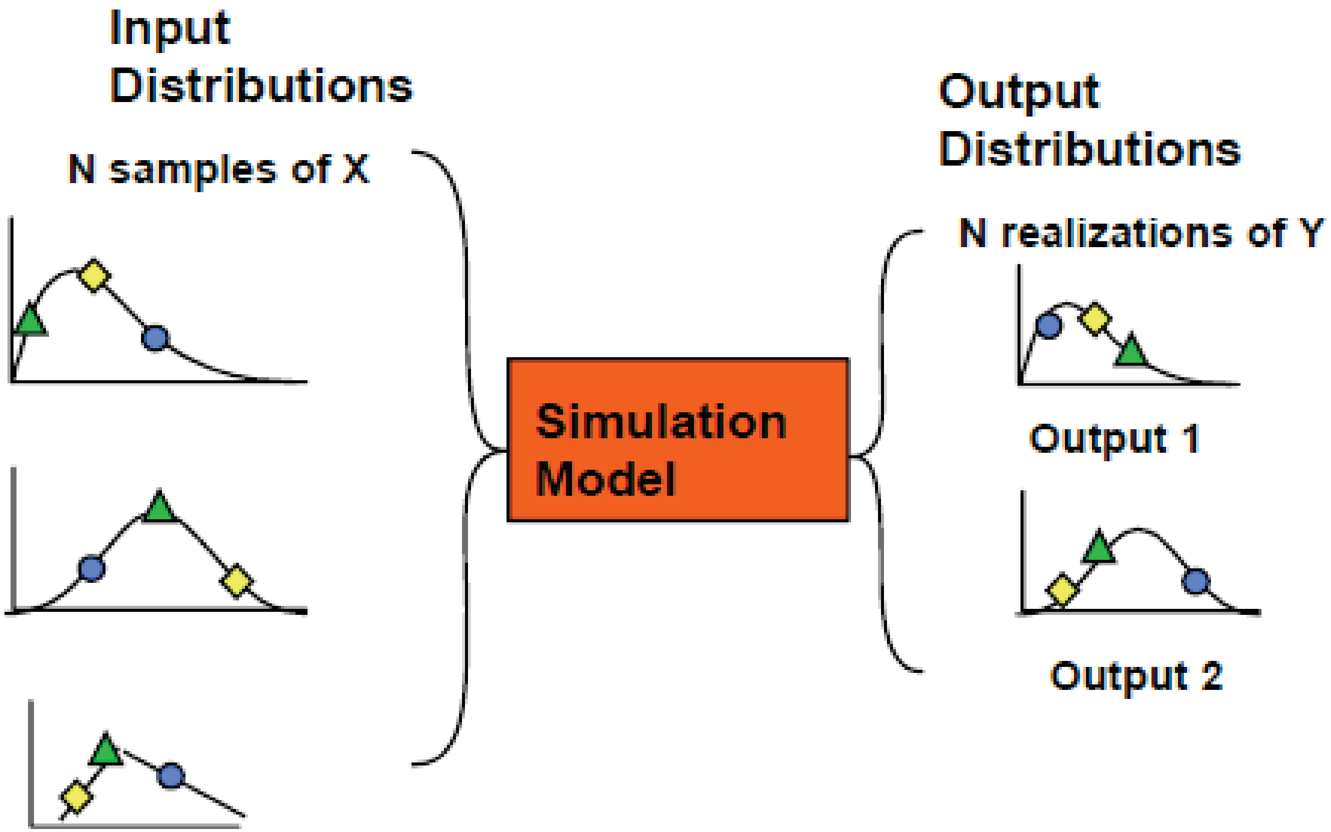
N Specific transient models and runs

Statistical analysis of the results

Licensing Case
(LC)

STATISTICAL UNCERTAINTY ANALYSIS METHOD

> PROPAGATION OF INPUT PARAMETER UNCERTAINTIES



STATISTICAL UNCERTAINTY ANALYSIS METHOD

> NON-PARAMETRIC ORDER STATISTICS

- Determination of the **Wilks' estimator (1st order)** with minimum number of calculations (N)
  - one-sided tolerance limit

$$1 - \gamma^N = \beta$$

  - double-sided tolerance limits

$$1 - \gamma^N - N\,(1-\gamma)\,\gamma^{N-1} = \beta$$

$\beta \times 100$ = confidence level (%)

$\gamma \times 100$ = probability (quantile) (%)

E.g., $\gamma = \beta = 0.95$: N=59 (single-sided), and 93 (double-sided)

STATISTICAL UNCERTAINTY ANALYSIS METHOD

> NON-PARAMETRIC ORDER STATISTICS & BOOTSTRAP

- Determination of the **Guba&Makai's estimators**
  - Order-statistic (kth rank)

$$\beta = (1-\gamma)^N + \sum_{j=1}^{N-k} \left\{ \prod_{l=1}^{j} \frac{N-l+1}{l} \gamma^j (1-\gamma)^{N-j} \right\}$$

  - Truncated order-statistic (tth rank)

$$t = \{INTEGER(N * 0.95) + 2\}$$

  - Estimators determined from the ordered outputs sample :

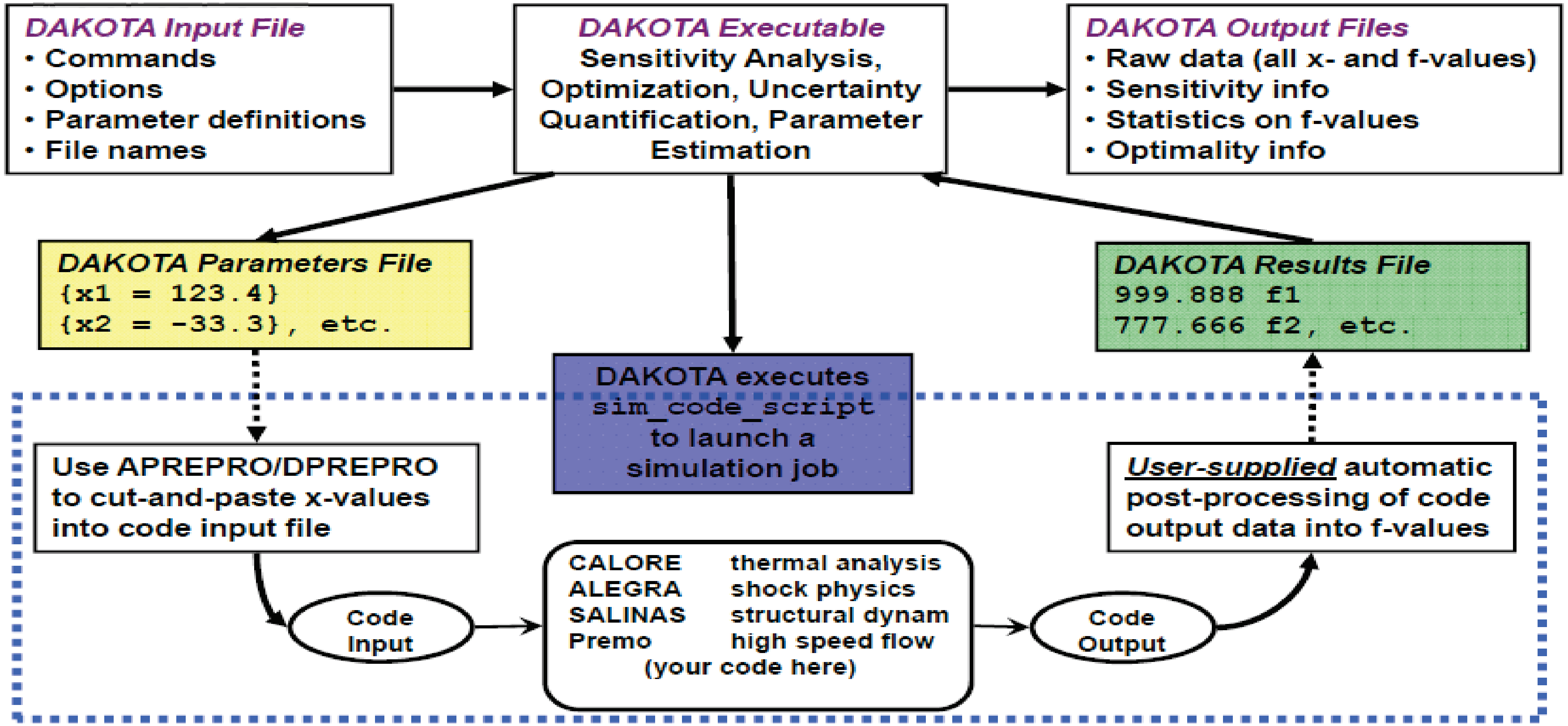
$y(1)<\ldots<y(i^{th})<\ldots<y(t^{th})<\ldots<y(k^{th})<\ldots<y(N^{th})$.

- Other statistics based on **bootstrap methods**
  - kernel smoothing;
  - non parametric density estimation.

STATISTICAL UNCERTAINTY ANALYSIS TOOL

> THE DAKOTA CODE

- DAKOTA = Design Analysis Kit for Optimization and Terascale



DAKOTA Input File
- Commands
- Options
- Parameter definitions
- File names

DAKOTA Executable
Sensitivity Analysis, Optimization, Uncertainty Quantification, Parameter Estimation

DAKOTA Output Files
- Raw data (all x- and f-values)
- Sensitivity info
- Statistics on f-values
- Optimality info

DAKOTA Parameters File
`{x1 = 123.4}`
`{x2 = -33.3}, etc.`

DAKOTA executes `sim_code_script` to launch a simulation job

DAKOTA Results File
`999.888 f1`
`777.666 f2, etc.`

Use APREPRO/DPREPRO to cut-and-paste x-values into code input file

Code Input

CALORE	thermal analysis
ALEGRA	shock physics
SALINAS	structural dynam
Premo	high speed flow

(your code here)

Code Output

__User-supplied__ automatic post-processing of code output data into f-values

Sandia National Laboratories

PRELIMINARY APPLICATIONS AND RESULTS

> THE OECD/WGFS RIA FUEL CODES BENCHMARK

- Objective
  - to assess the ability of fuel rod codes to reproduce the results from Reactivity Initiated Accidents (RIA) experiments performed in NSRR and CABRI test reactors…

    with a certain degree of adequacy.
- Uncertainty/sensitivity analysis needed
  - To consider the impact of the uncertainties
  - To provide certain confidence

→ In line with the ongoing BEMUSE project and the UAM project

PRELIMINARY APPLICATIONS AND RESULTS

> THE OECD/WGFS RIA FUEL CODES BENCHMARK

- The CIP3-1 blind test case
  - RIA test to be performed in the CABRI reactor
  - High burnup PWR fuel rod
- FRAPCON3.4/FRAPTRAN1.4 Simulation of CIP3-1
  - « Best Estimate » calculation
    - **specified nominal values**
    - **default model options**

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PRELIMINARY APPLICATIONS AND RESULTS

> THE OECD/WGFS RIA FUEL CODES BENCHMARK

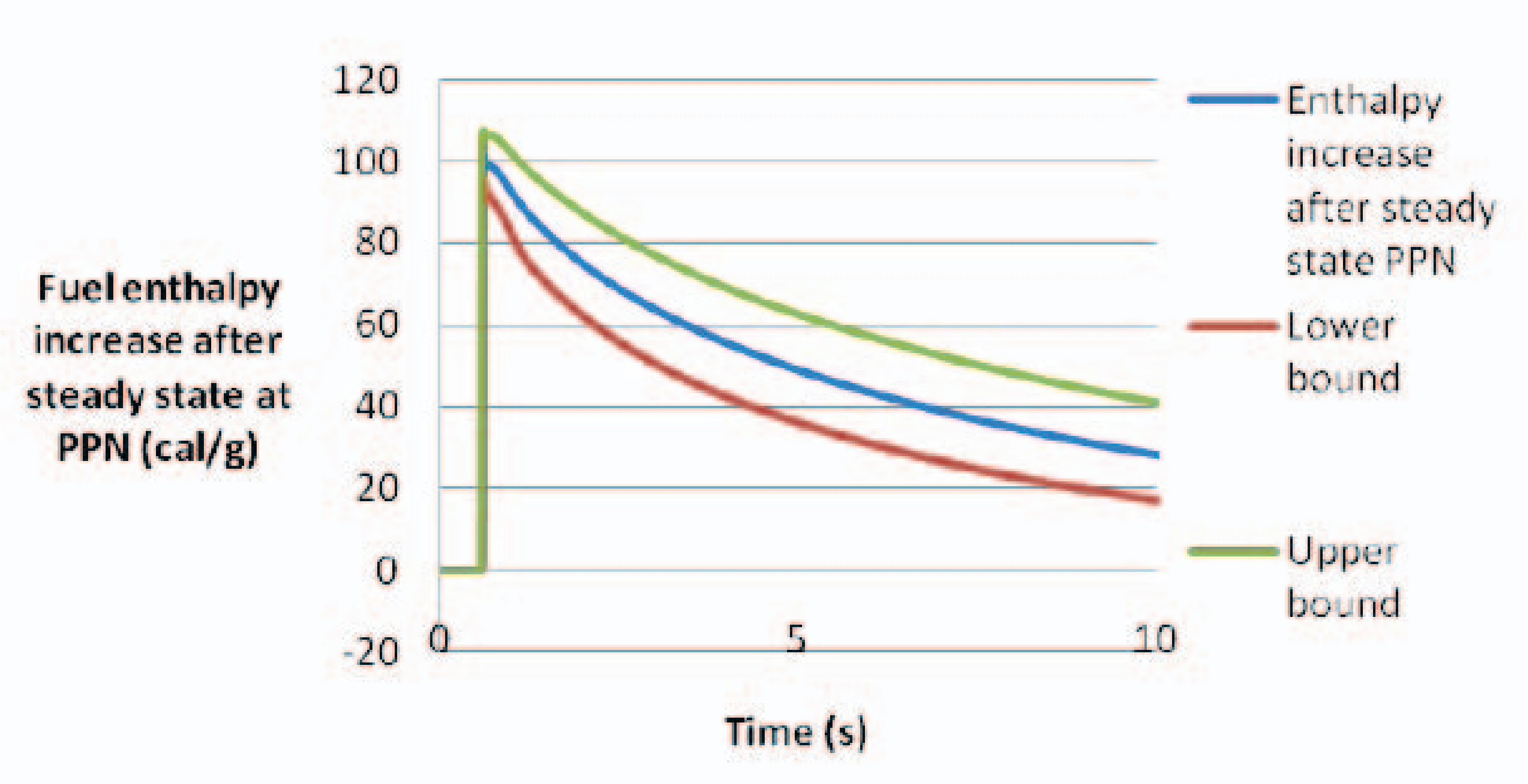
- Identification and definition of input uncertain parameters

Input uncertainty parameter	**Mean**	**Standard deviation**	**Lower bound**	**Upper bound**	**Distribution**
Thermal conductivity model	0	1	-2	2	Normal
Thermal expansion model	0	1	-2	2	Normal
Fission gas release model	0	1	-2	2	Normal
Fuel swelling model	0	1	-2	2	Normal
Cladding creep model	0	1	-2	2	Normal
Cladding corrosion model	0	1	-2	2	Normal
Cladding hydrogen uptake model	0	1	-2	2	Normal
Multiplicative factor on the temperature history during base irradiation	1	0,00355	0,9929	1,0071	Normal
Multiplicative factor on the power history during base irradiation	1	0,02	0,96	1,04	Normal
Multiplicative factor on the power pulse	0,92976	0,0186	0,89257	0,96695	Normal
Coolant inlet enthalpy (J/kg) during the transient	1232080	5080	1221920	1242240	Normal
Cladding outside diameter (m)	0,0095	0,000019	0,009462	0,009538	Normal
Cladding inside diameter (m)	0,008357	0,000019	0,008319	0,008395	Normal
Dish radius (m)	0,002475	0,0000625	0,00235	0,0026	Normal
Fuel density (%)	95,5	0,75	94	96,5	Normal
Pellet diameter (m)	0,008192	0,000006	0,00818	0,008204	Normal
Cladding roughness (µm)	0,6355	0,31725	0,001	1,27	Normal
Fuel roughness (µm)	1,6005	0,79975	0,001	3,2	Normal
Cold plenum length during base irradiation (m)	0,029531	0,000884	0,0278	0,0301	Normal

PRELIMINARY APPLICATIONS AND RESULTS

> THE OECD/WGFS RIA FUEL CODES BENCHMARK

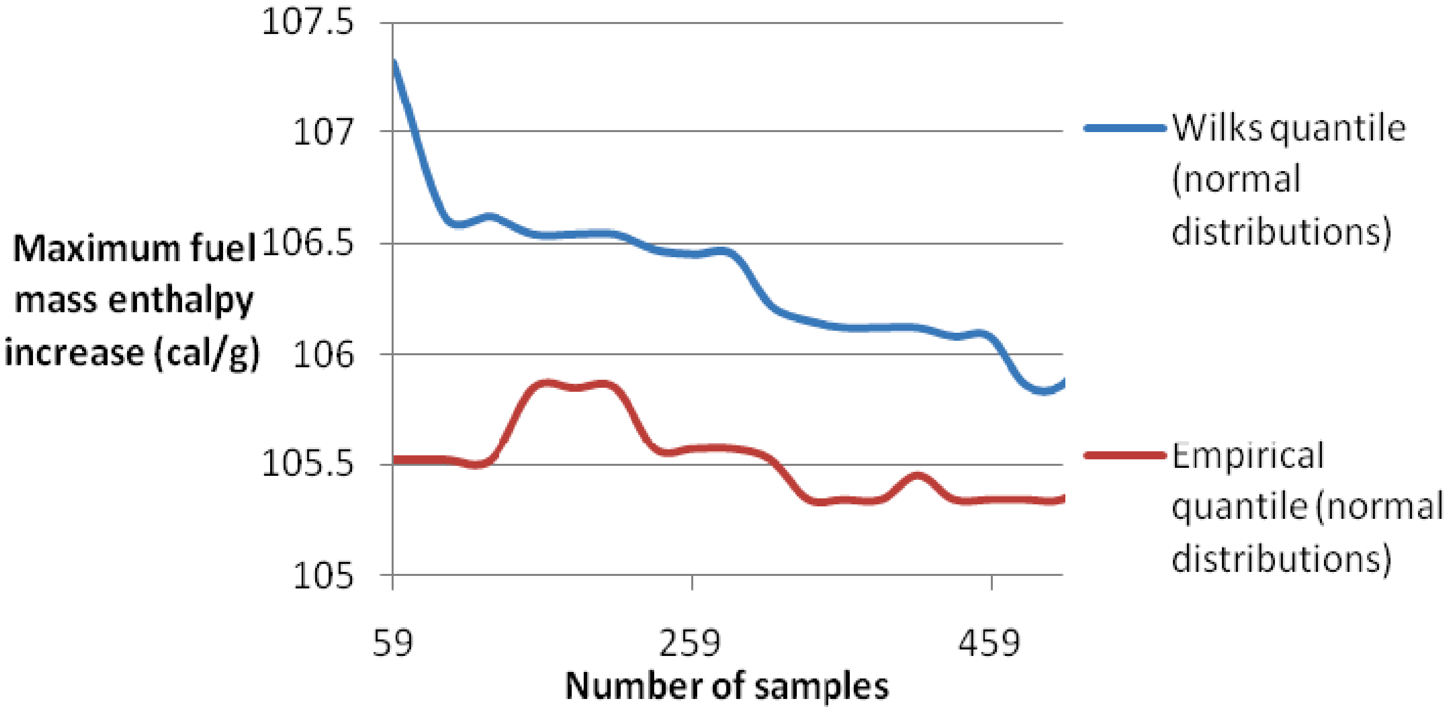
- The Upper/Lower Bound Values (Double-sided , N=93)



PRELIMINARY APPLICATIONS AND RESULTS

> THE OECD/WGFS RIA FUEL CODES BENCHMARK

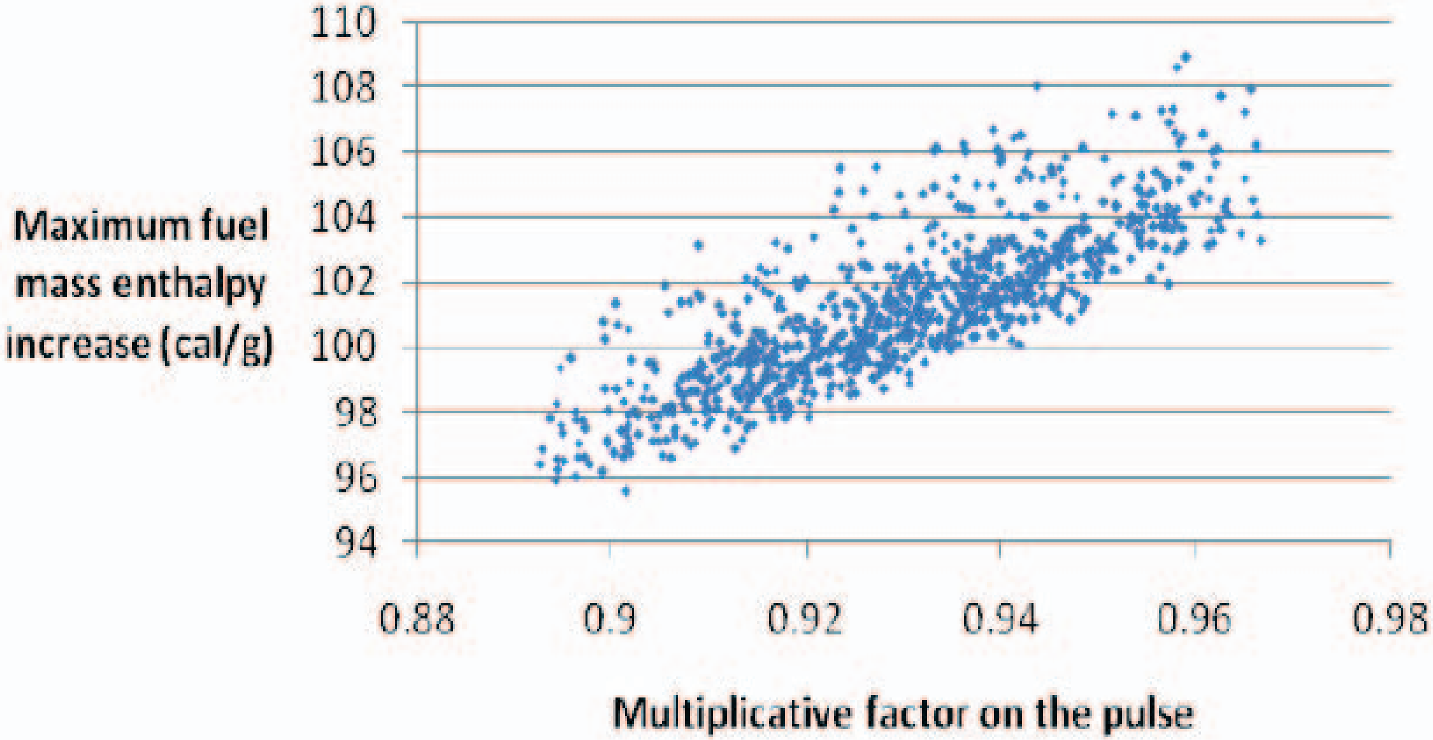
- The Wilks' estimator → empirical quantile with larger N



PRELIMINARY APPLICATIONS AND RESULTS

> THE OECD/WGFS RIA FUEL CODES BENCHMARK

- Sensitivity study → importance of uncertainty parameters



PRELIMINARY APPLICATIONS AND RESULTS

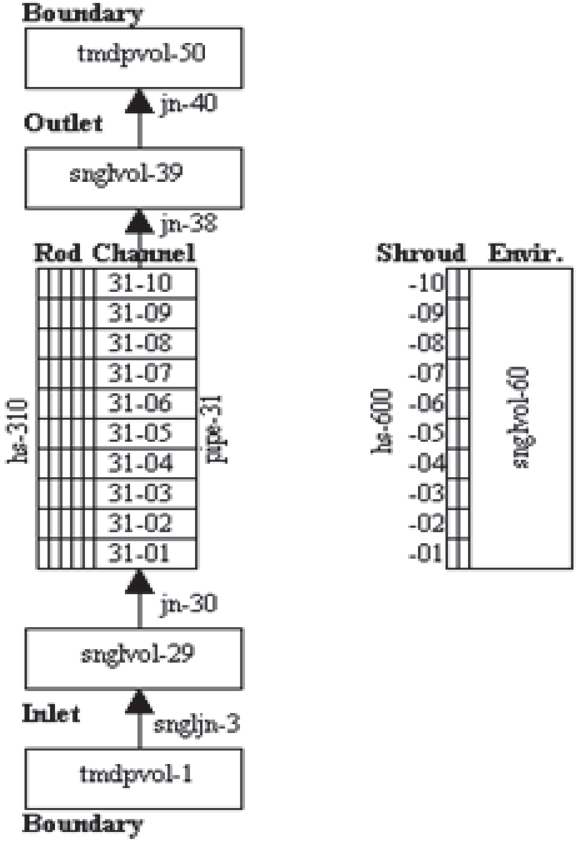
> RELAP5 SIMULATION OF THTF TESTS

- Objective
  - to compare the robustness of different statistical uncertainty analysis methods
    - **Order statistic method for Guba&Makai's estimators: N = 59, 153, 1000, 4000.**
    - **Bootstrap methods with different estimators: n= 100 or 200 resamples (N=153).**
- RELAP5/MOD3.3 simulation of a THTF test
  - Sampling 4 most significant uncertain input parameters
  - The output parameter of interest: "Temperature at the Cladding".

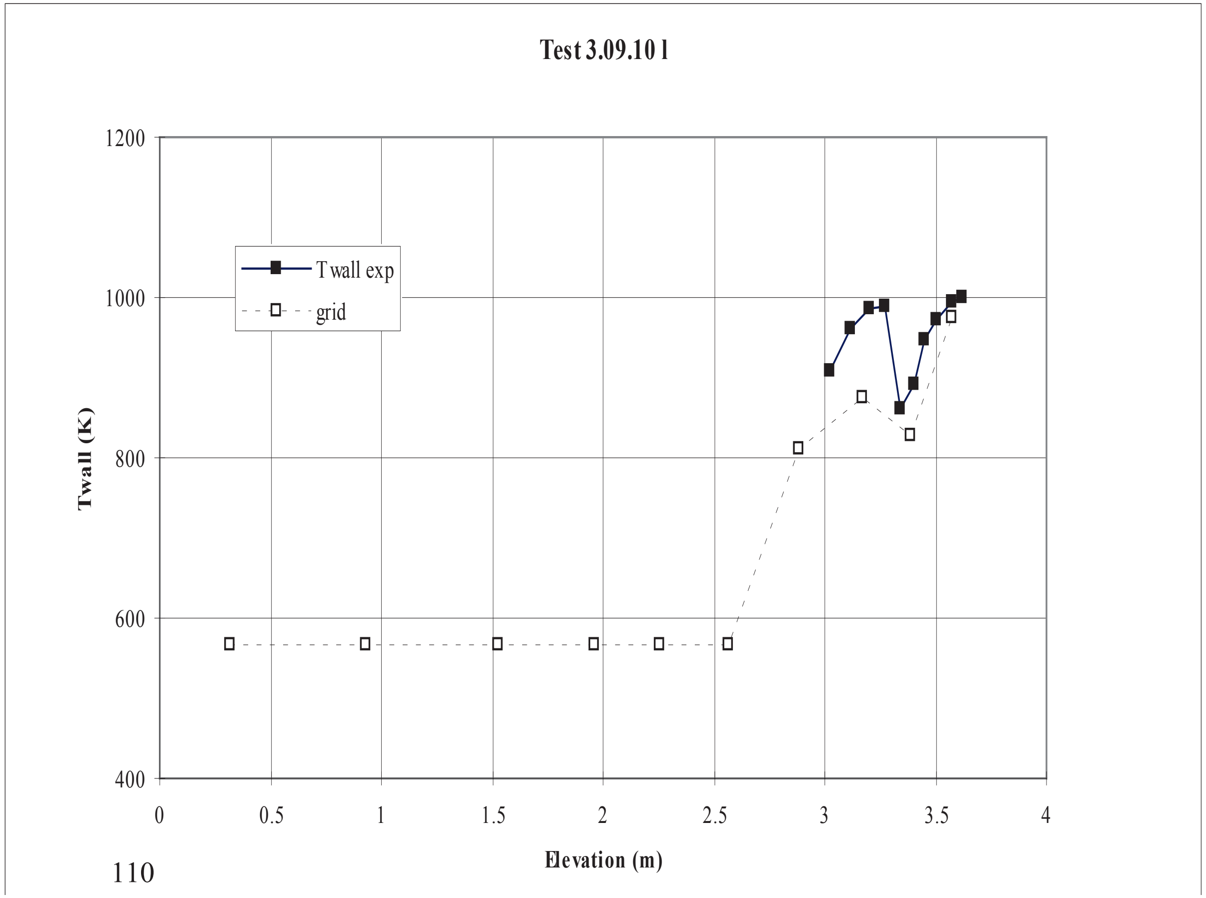
PRELIMINARY APPLICATIONS AND RESULTS

> RELAP5 SIMULATION OF THTF TESTS

- RELAP5/Mod3.3 model



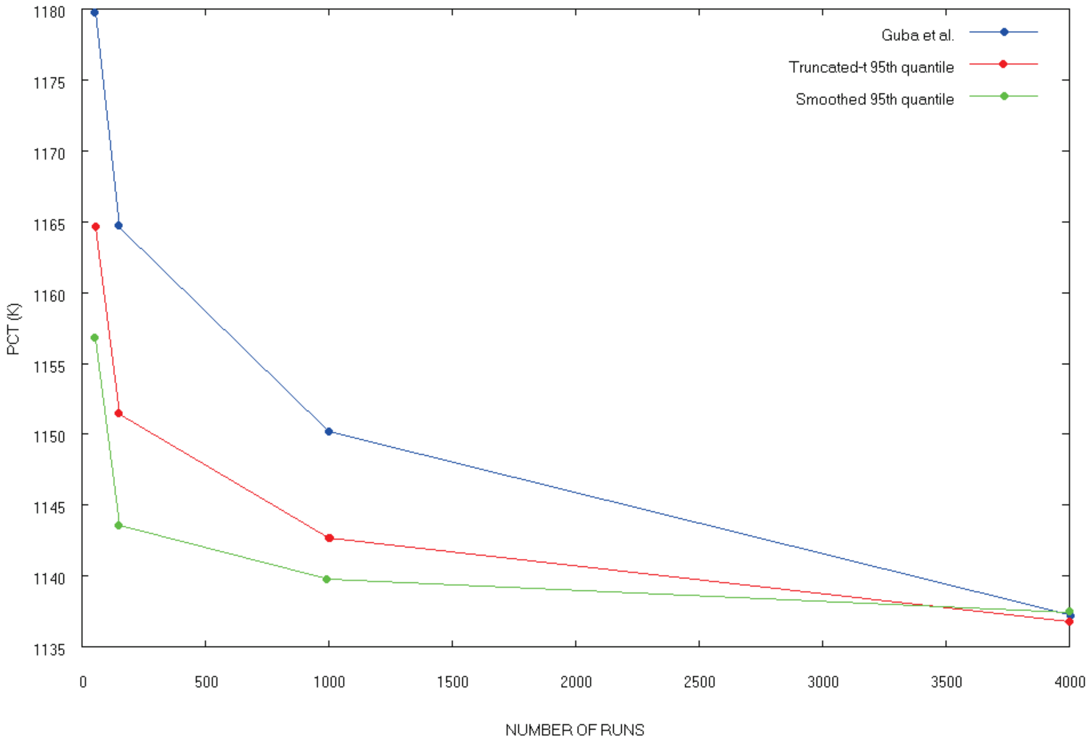
- Results for test 3.09.10 L



PRELIMINARY APPLICATIONS AND RESULTS

> RELAP5 SIMULATION OF THTF TESTS

- Estimation of the 95th quantile by order statistics

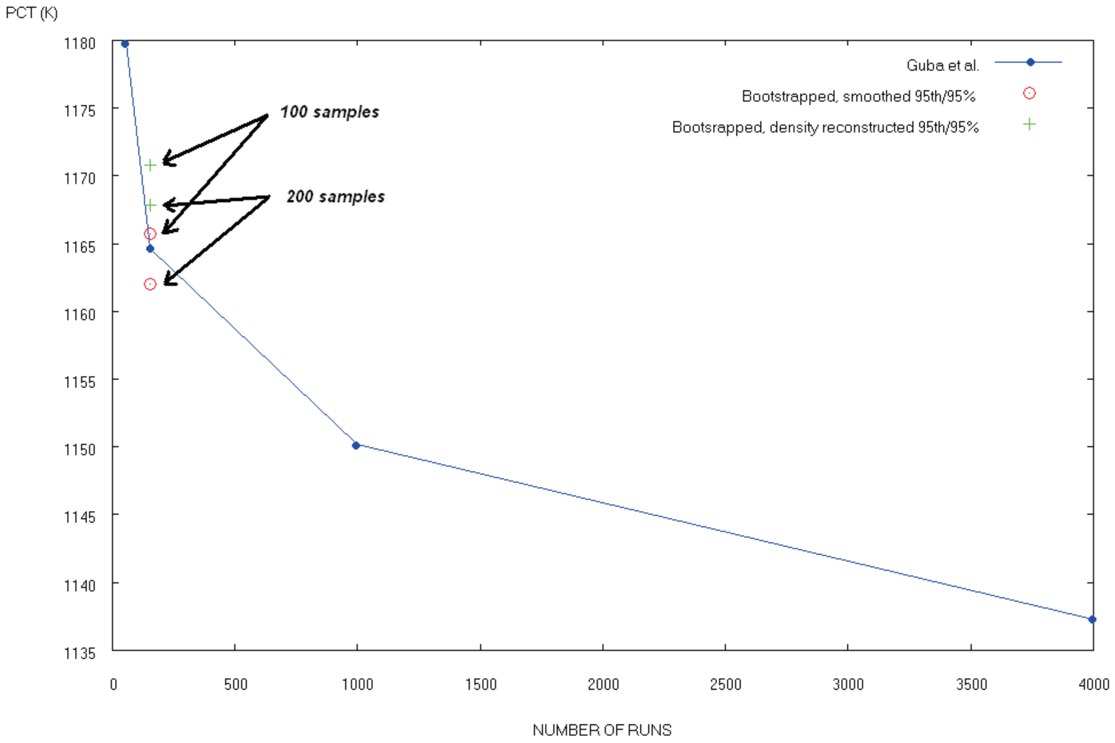


N → 4000 : Guba&Makai's kth rank estimator → the truncated tth rank estimator

PRELIMINARY APPLICATIONS AND RESULTS

> RELAP5 SIMULATION OF THTF TESTS

- Comparison of two different bootstrap quantile estimators



The percentile smoothing L-estimator → Guba&Makai's k^{th} rank estimator

CONCLUSIONS AND PERSPECTIVES

- The TE best estimate plus statistical uncertainty analysis method (BESUAM) proposed
  - Based on TE current multi-physics code package (PANTHER, COBRA, RELAP, FRAPCON/FRAPTRAN)
  - Extensive Verification & Validation (V&V) needed to quantify the code uncertainties (UQ) → future work
  - Use of non-parametric order statistic and/or bootstrap uncertainty analysis methods
  - With a robust sensitivity and uncertainty analysis tool (DAKOTA)
- Final objective
  - apply to reloads fuel safety evaluation and safety analysis

CONCLUSIONS AND PERSPECTIVES

- The non-parametric order statistic methods are simple, transparent, robust and flexible
  - Will be further applied to the UAM benchmark phase II on
    - **fuel physics (II-1),**
    - **neutronics (II-2) and**
    - **bundle thermal hydraulics (II-3)**
- Further work/needs
  - to improve and qualify the multi-physics code package and input models → *V&V*
  - to identify, quantify or define the ranges and distributions of all relevant uncertainty attributors → *PREMIUM?*
  - to practically separate and treat aleratory and epistemic uncertainties → *nested sampling (or second-order probability) method?*
  - to ensure the compatibility with the current reloads safety evaluation method based on nuclear key safety parameters

OECD/CSNI Workshop on
Best Estimate Methods and Uncertainty Evaluations
UPC & CSN, Barcelona, Spain, November 16-18, 2011

Experiences in KINS on Best Estimate Calculation with Uncertainty Evaluations

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Contents

- ☐ Introduction
- ☐ KINS-Realistic Evaluation Method
- ☐ Application to LBLOCA of Actual Plants
- ☐ Results and Findings

1. Introduction

- **BEPU methodology in Korean Industries**
  - Continuous efforts in industries and research institutes since late 1980’s
  - KEPCO Realistic Evaluation Method (KREM) firstly approved at 2002 (started from 1997)
  - KREM has been expanded to Various types of ECCS, Upper Plenum Injection (UPI), Direct Vessel Injection (DVI), etc.
- **BEPU methodology in KINS**
  - KINS-REM developed for regulatory auditing calculation in parallel (originally similar to CSAU method)
  - Improved by participating OECD BEMUSE Program
  - Recently, MARS-KS code implemented

1. Introduction

☐ Difficulties

- How enough best-estimate (code and modeling)
- How to deal with the uncertainty for the specific phenomena

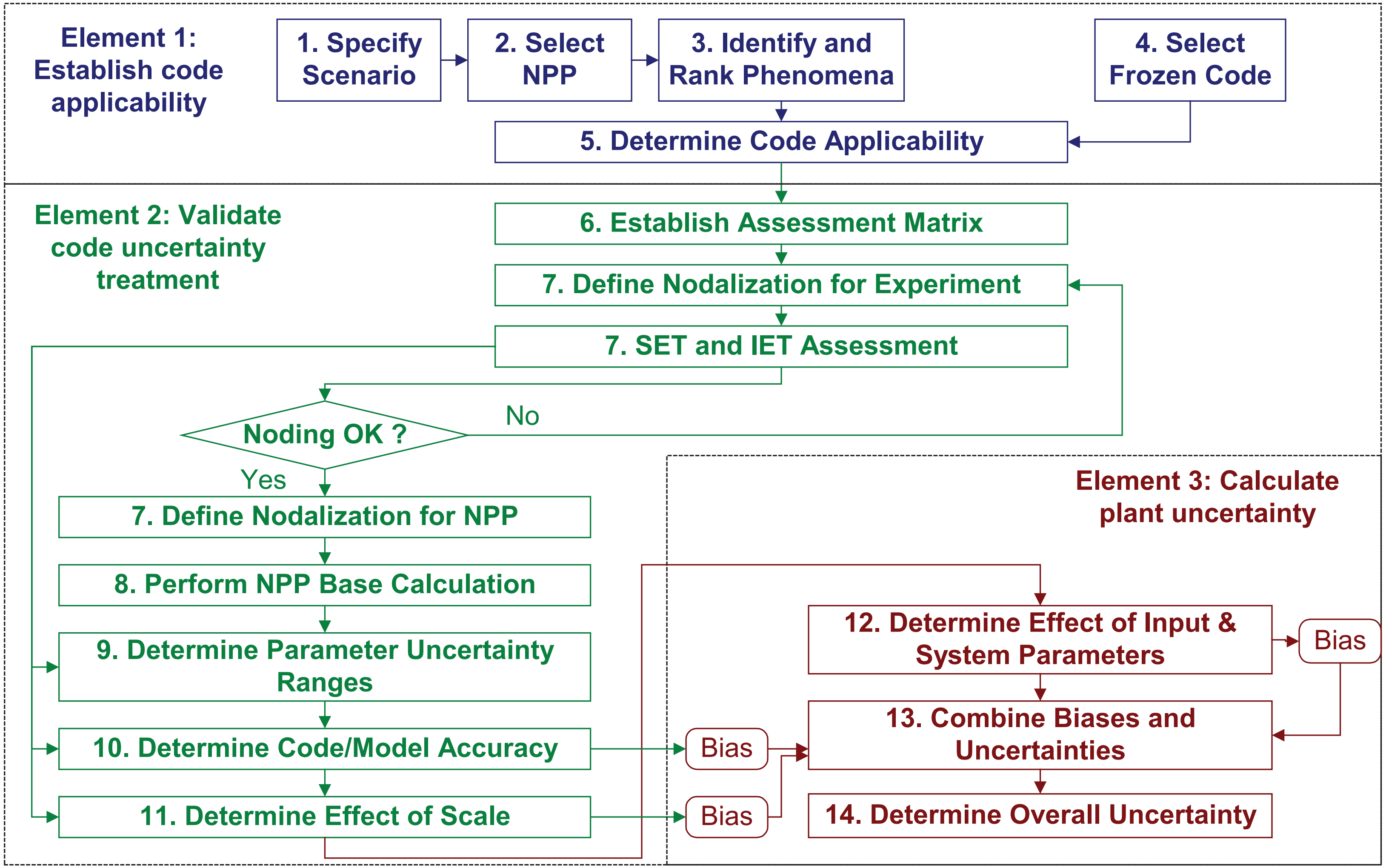
☐ Status in Korea

- New system thermal-hydraulic code (SPACE) being developed → to be reviewed by KINS
- Licensing process ongoing for continuous operation of Wolsong Unit 1 →
- Other complicated issues

☐ Objectives

- Discuss some aspects of BEPU based on KINS experience, listen comments, and look at the path forward

2. KINS-REM



Element 1: Establish code applicability

1. Specify Scenario

2. Select NPP

3. Identify and Rank Phenomena

4. Select Frozen Code

5. Determine Code Applicability

Element 2: Validate code uncertainty treatment

6. Establish Assessment Matrix

7. Define Nodalization for Experiment

7. SET and IET Assessment

Noding OK ?

No

Yes

7. Define Nodalization for NPP

8. Perform NPP Base Calculation

9. Determine Parameter Uncertainty Ranges

10. Determine Code/Model Accuracy

11. Determine Effect of Scale

Element 3: Calculate plant uncertainty

12. Determine Effect of Input & System Parameters

Bias

13. Combine Biases and Uncertainties

Bias

Bias

14. Determine Overall Uncertainty

2. KINS-REM

Principle

- All source of uncertainties considered (statistically or in terms of engineering bias)
- Uncertainty due to modeling minimized during the SET/IET assessment and not considered in plant calculation

PCT and PCT Uncertainty

$$PCT_{final} \cong PCT_{95/95} + B_{SET} + B_{IET} + B_{PLANT} + B_{SCALE}$$

- PCT_{final} :
  - Final PCT accounting all uncertainties and biases
- $PCT_{95/95}$:
  - PCT at 95% probability and 95% confidence level
  - the 3rd maximum of the PCT's calculated by the N code runs implementing values sampled from the range and distribution of all uncertainty parameters

2. KINS-REM

- N=124 : determined from the 3rd order Wilk's formula

$$q = 1 - [p^N + N(1-p)p^{N-1} + \frac{1}{2}N(N-1)(1-p)^2 p^{N-2}]$$

- $B_{SET/IET}$:
  - Impact on PCT due to SET/IET bias (Step 10)
  - Steam binding bias considered currently
- B_{SCALE} :
  - Impact on PCT due to scaling problem (Step 11)
- B_{PLANT} :
  - Impact on PCT or equivalent due to the plant parameter unconsidered in PCT95/95 calculation (Step 12)

3. Application to LBLOCA of Actual Plants

Cases of auditing calculation using KINS-REM

Plant	Design	ECCS *	Application Type	Code
Kori 1	Westinghouse 2-loop	CLI/UPI[1]	Licensing renewal	RELAP5/MOD3.3
Kori 2	Westinghouse 2-loop	CLI/DVI[2]	Fuel design change	RELAP5/MOD3.3
YGN 3,4	Korean OPR1000	CLI/CLI	Methodology change	RELAP5/MOD3.3
SKN 3,4	Korean APR1400	DVI/DVI[3]	Construction permit	RELAP5/MOD3.3 MARS-KS
UCN 1,2	Westinghouse 3-loop	CLI/CLI	SG Replacement	MARS-KS
Wolsong 1	CANDU-6 PHWR	Header[4]	Continuous Operation*	MARS-KS

- Note 1: UPI: Upper Plenum Injection for Low Pressure Safety Injection

  2: DVI: Direct Vessel Injection (Elevation of Nozzle ~ Cold Leg)

  3: DVI: Direct Vessel Injection (Elevation of Nozzle ~ 2 m higher than Cold Leg)

  4: Injection to Inlet/Outlet Headers (High/Medium/Low Stage Injection)

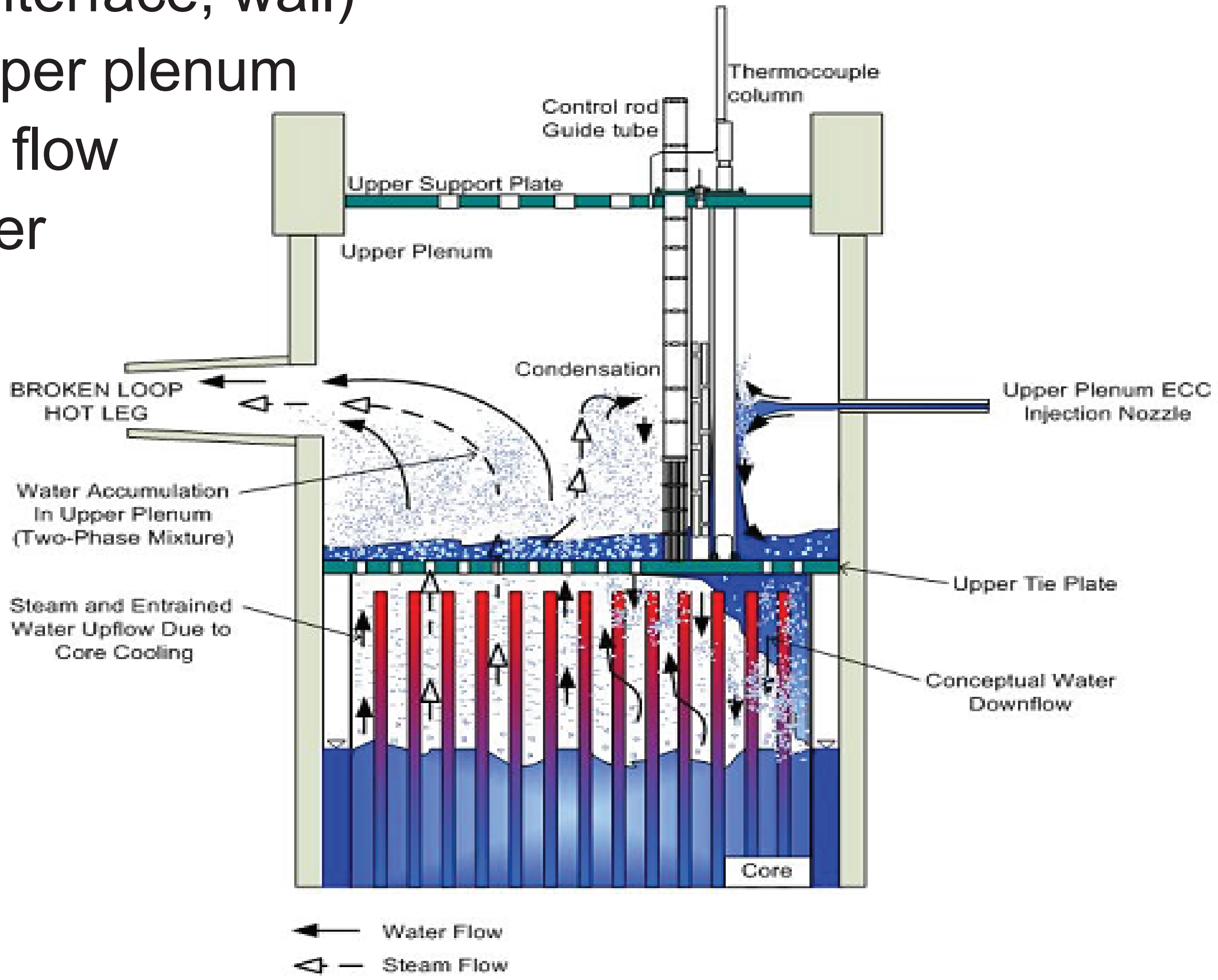
3.1 Thermal-hydraulic Phenomena

- **Approach**
  - Basis for selecting the Uncertainty Variables
  - Phenomena identification for sub-phases in scenario (Blowdown, Refill & Reflood)
  - Some phenomena considered for all sub-phases (i.e., Gap conductance)
  - Blowdown phenomena
    - Fuel stored energy, Break flow, Depressurization, Flashing, Core heat transfer, etc
    - Almost identical regardless of ECCS type
  - Refill phenomena
    - ECC bypass, Rewet, Core heat transfer, etc
    - Strongly dependent on reactor vessel, ECCS type
  - Reflood phenomena
    - Difference by ECCS design (CLI, UPI, DVI)

3.1 Thermal-hydraulic Phenomena

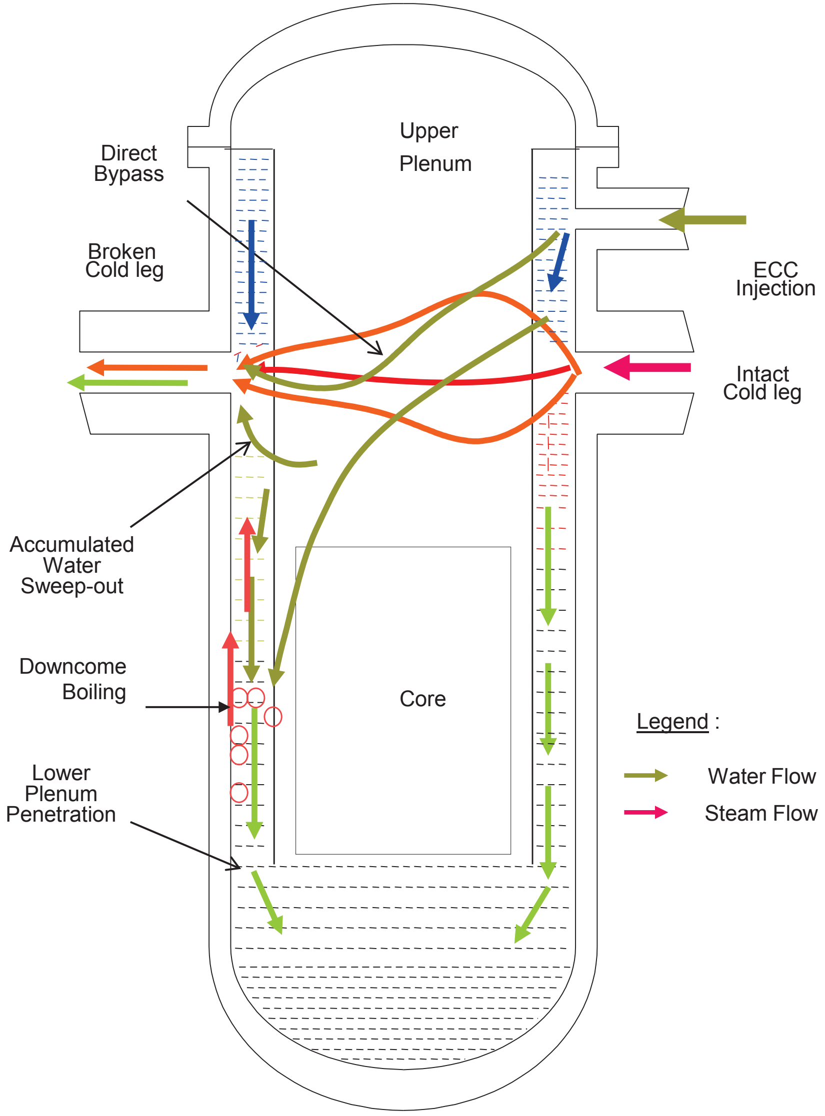
□ Reflood phenomena (UPI)

- Condensation (interface, wall)
- Water pool in upper plenum
- Local downward flow
- Core heat transfer



3.1 Thermal-hydraulic Phenomena

- **Reflood phenomena (DVI)**
  - ECC bypass
    - Direct bypass
    - Downcomer Boiling
    - Sweepout
    - Lower plenum penetration
  - Core heat transfer
  - Steam binding



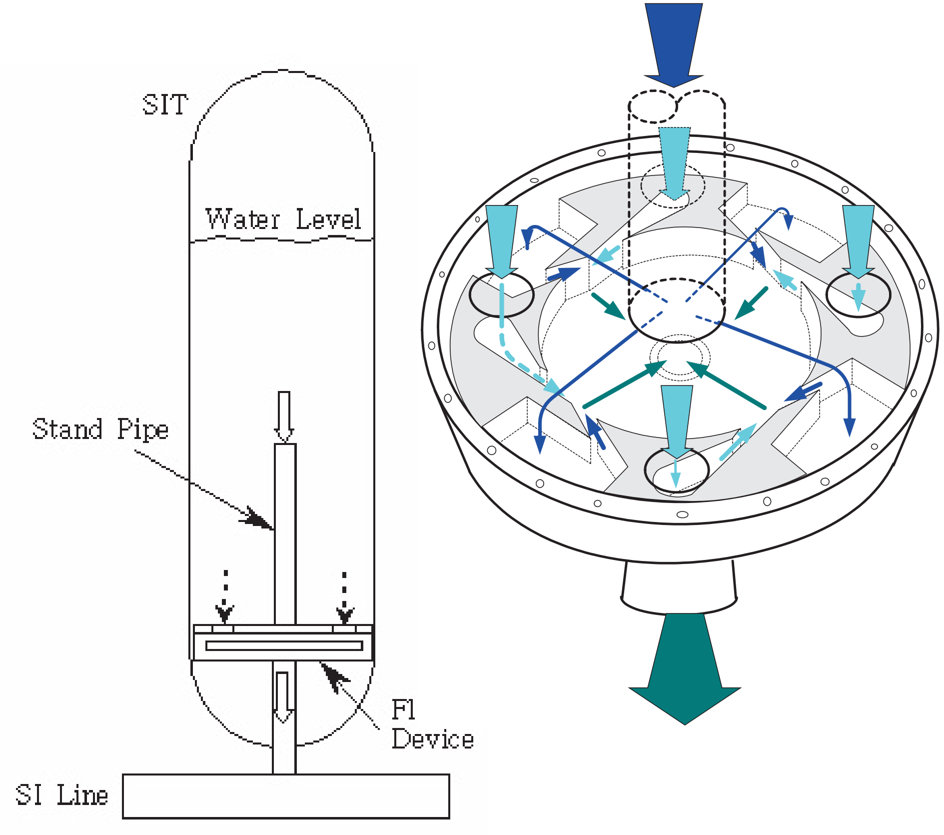
3.1 Thermal-hydraulic Tests

Facility (Scale)	Experiment	Phase of Interest	Availability
LOFT (1/60)	L2-2, L2-3, L205, LP-LB-1, LP-02-6	Full Transient	O
Semiscale (1/1600)	S-04-5, S-06-3, S-02-8, S-07-4	Full Transient	O
PKL (1/145)	K9	Refill/Reflood	x
LOBI (1/700)	A1-4R	Blowdown/Refill	x
SCTF (1/21)	S1-13, S2-08, S3-10	Reflood	x
CCTF (1/21)	C1-06, C1-16, C2-06, C2-15	Reflood	x
	C2-AS1, C2-16, C2-13	***Reflood (UPI)***	***O***
THTF (1/500)	154R, 153	Blowdown	O
FLECHT-SEASET (1/1, Core only)	31504,	Reflood	O
	30518	Reflood	O
UPTF (1/1, Reactor Vessel)	Test 2, Test 17, Test 10	Refill/Reflood/Steam Binding	x
	Test 20	***Reflood (UPI)***	O
	Test 27	Refill/Reflood	x
	Test 21A	Refill /ECC Delivery (DVI)	O
	Test 21D	Reflood (DVI)	O

3.1 Thermal-hydraulic Phenomena

◻ ECCS Design Features specific to Shin-Kori 3 & 4

- Fluidic Device to extend the injection time (high discharge phase and low discharge phase)



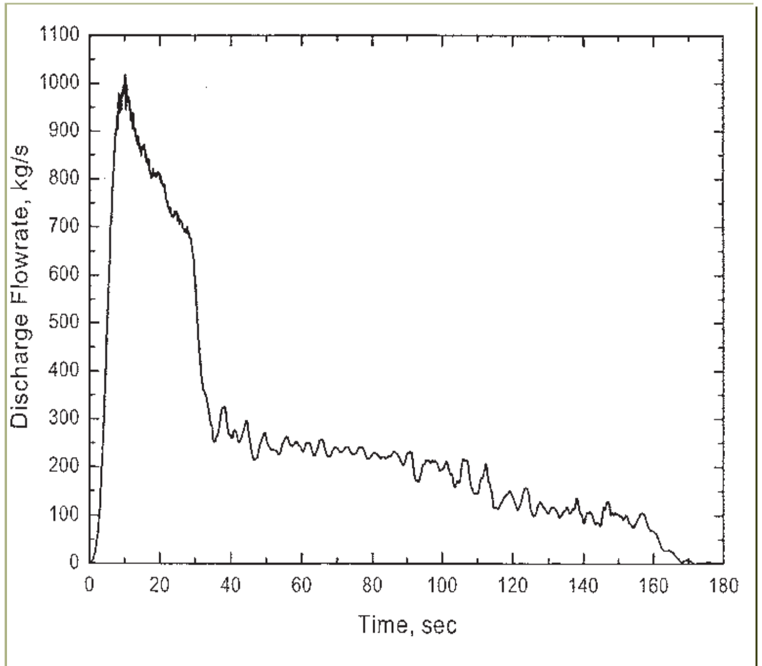


Fig. 5 Discharge Flowrate from SIT (Test-II(b)-C-H-1a)

3.2 Uncertainty Parameters

Phenomena /sub-phenomena	Phase /component	Models/ parameters	Uncertainty Treatment	Distribution/ Range
Fuel stored energy	Blowdown/ Fuel	Gap conductance Peaking factor Dittus-Boelter correlation Fuel conductivity Initial core power Burst temp/strain	Statistic Upper limit Statistic Statistic Statistic Not considered	0.4~1.5/N 0.606~1.39/N 0.847~1.153/U 0.98~1.02/N
Clad oxidation	Reflood/Fuel	Cathcart-Powell Cor.	Not considered	
Decay heat	Reflood/Fuel	ANS decay heat model	Statistic	0.934~1.066/N
Break flow	All /Break	Discharge coefficient	Statistic	0.729~1.165/N
2-ϕ pump performance	Reflood/Loop	2-ϕ head multiplier 2-ϕ torque multiplier	Statistic Statistic	0.0~1.0/U 0.0~1.0/U
ECC Bypass **Condensation** **Entrainment** **CCFL** **Hot wall effect** **Sweepout** **Multi-D flow**	Reflood/ Downcomer	 Wall condensation model Entrainment model CCFL model Wall-to-fluid HTC Interfacial drag model Crossflow resistance	Evaluate conservatism based on experiment data and add bias if needed	
Void generation	Blowdown/ Core	Boiling model	Not considered	
2-ϕ frictional pressure drop	Reflood/Loop	2-ϕ friction multiplier	Not considered	

3.2 Uncertainty Parameters

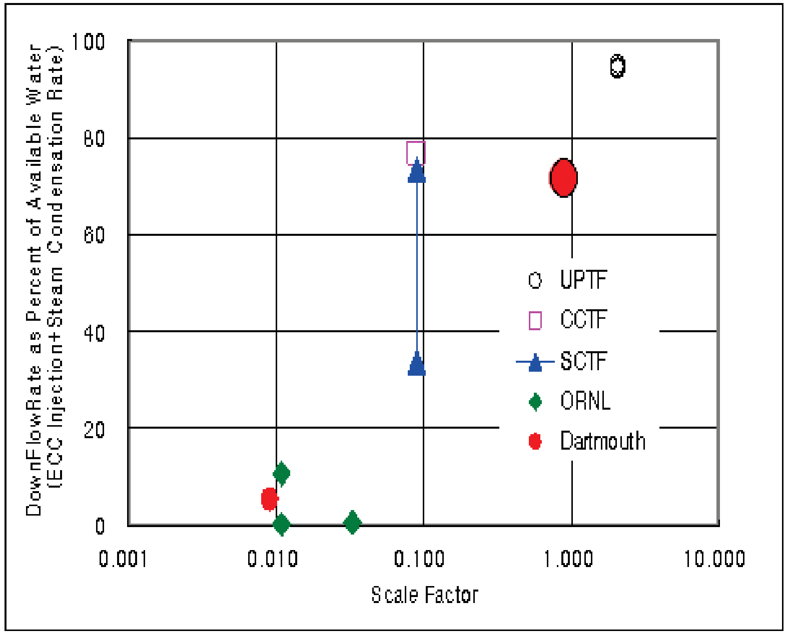
Phenomena /sub-phenomena	**Phase /component**	**Models/ parameters**	**Uncertainty Treatment**	**Distribution/ Range**
Core heat transfer	All /Core			
CHF		Groeneveld lookup table	Statistic	0.17~1.8/N
Rewet		Rewet criteria	Not considered	
Transition boiling		Transition boiling HTC	Statistic	0.54~1.46/N
Film boiling		Bromley correlation	Statistic	0.428~1.58/N
Nucleate boiling		Chen correlation	Statistic	0.53~1.46/N
Reflood htc		Reflood htc package	Statistic	As above
Effect of non-condensible gas		Gas transport	Not considered	
Steam binding	Reflood/ U-tube, Core	Dittus Boelter correlation	Bias	
Containment pressure	Reflood/ Boundary	Boundary condition	Conservative input	
UPI phenomena **Pool formation** **local downward flow** **Condensation**	Reflood /Upper plenum	Momentum equation. flow regime map, etc. Interface condensation	Modeling pursuant to experiment, Bias if needed	
ECCS performance	Reflood/ ECCS	SIT pressure	Statistic	Lower and upper values from the plant Tech. Spec/U
		SIT temperature	Statistic	
		SIT water volume	Statistic	
		SIT flow resistance	Statistic	
		HPSI flow rates	Lower limit	
		HPSI temperature	Statistic	

3.3 Thermal-hydraulic Code

- **RELAP5/MOD3.3 & MARS-KS**
  - MARS-KS
    - Developed to keep the capability of RELAP5/MOD3 code and COBRA-TF code (same basic equations, numerical scheme and models/correlation)
    - Code modular structure and data storage management
    - 3-D hydrodynamic capability and Coupling capability with containment codes and 3-D neutronics code
    - Differences in reflood model
  - Predictability of Blowdown phenomena
    - Both codes predicted well or a little conservatively against the various SET and typical IET (LOFT, Semiscale, etc)

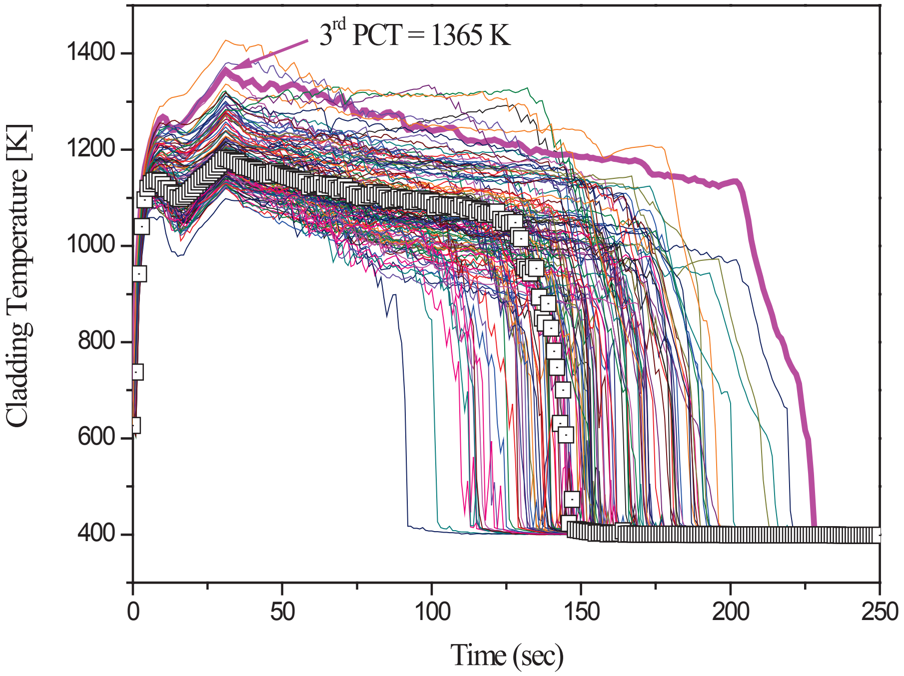
3.3 Thermal-hydraulic Code

- Predictability of Reflood phenomena
  - Direct code assessment calculation
    - Reflood with CLI ~ FLECHT-SEASET experiments
    - Reflood with UPI ~ UPTF, CCTF experiments
    - Reflood with DVI ~ UPTF, ATLAS experiments
  - Further improvement of MARS-KS code needed (ongoing)
- Modeling improvement
  - To match the code prediction with the experimental finding

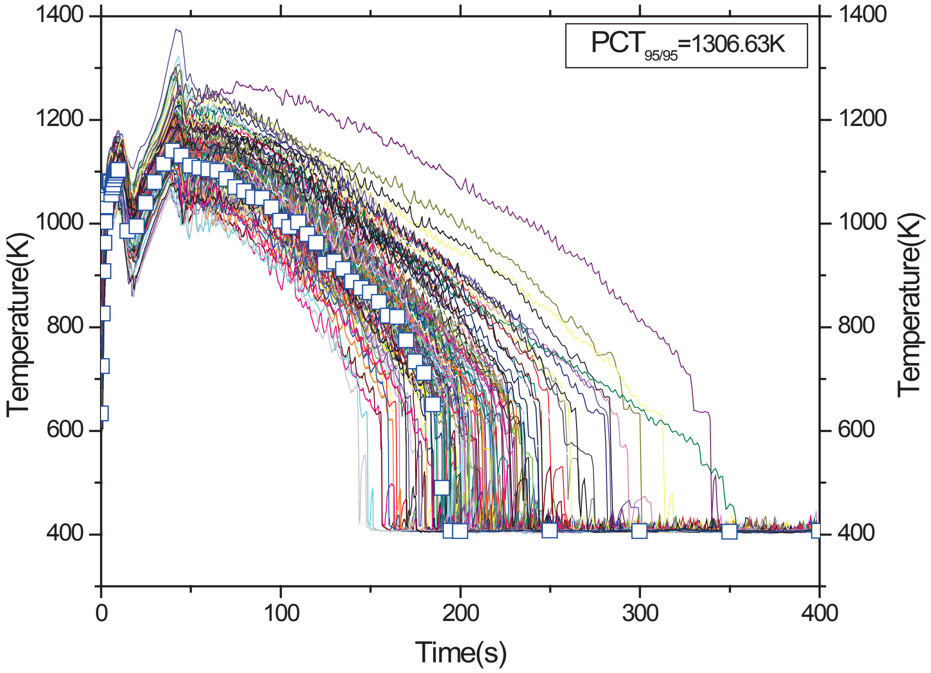


4. Results and Findings

□ Kori Unit 1 (UPI)



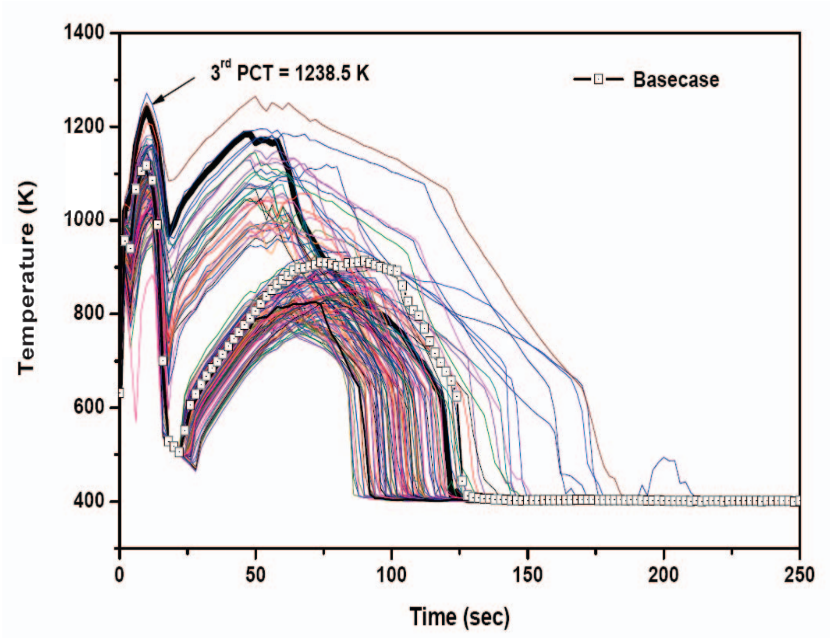
□ Kori Unit 2 (DVI)



4. Results and Findings

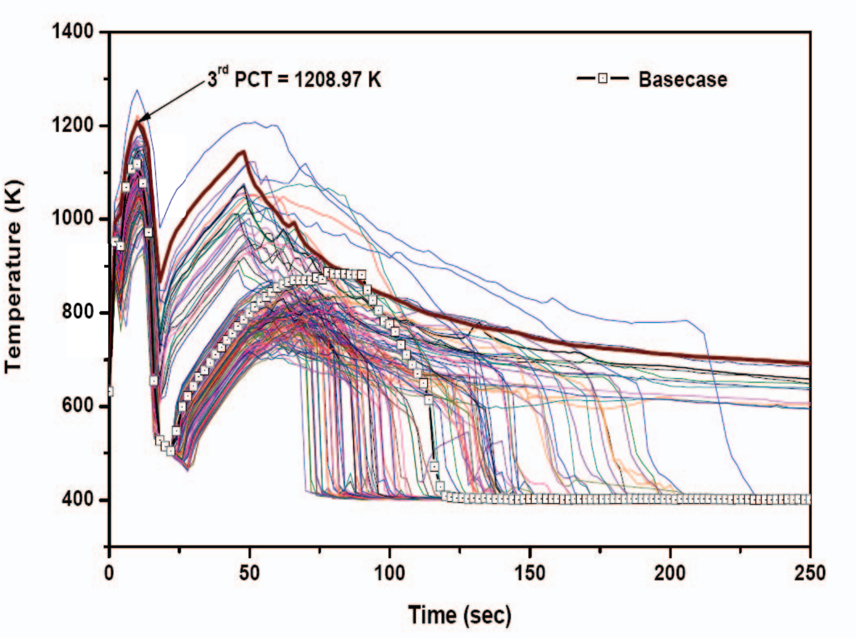
SKN Unit 3, 4 (DVI)

- RELAP5/MOD3.3



SKN Unit 3, 4 (DVI)

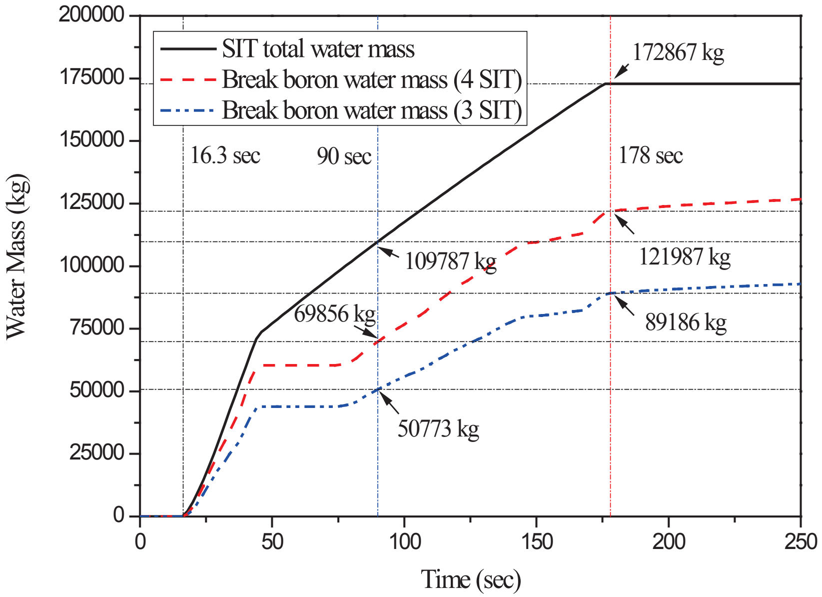
- MARS-KS

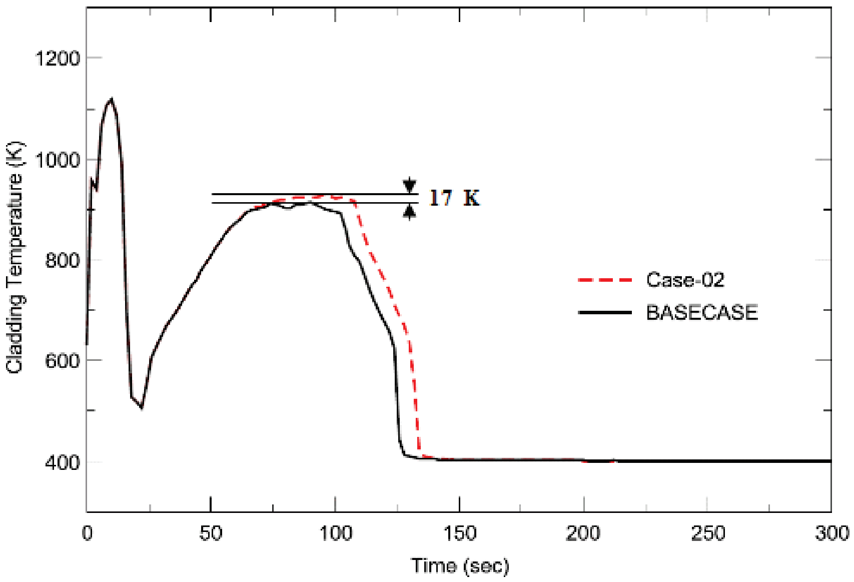


4. Results and Findings

◻ Additional Bias (SKN Unit 3, 4 (DVI))

- Calculation of UPTF test and MIDAS tests, ECCS bypass ratio over-predicted by both codes → no bias
- Finding of CCTF test, steam quality of SG outlet plenum during steam binding phenomena under-predicted → bias to match the test result





4. Results and Findings

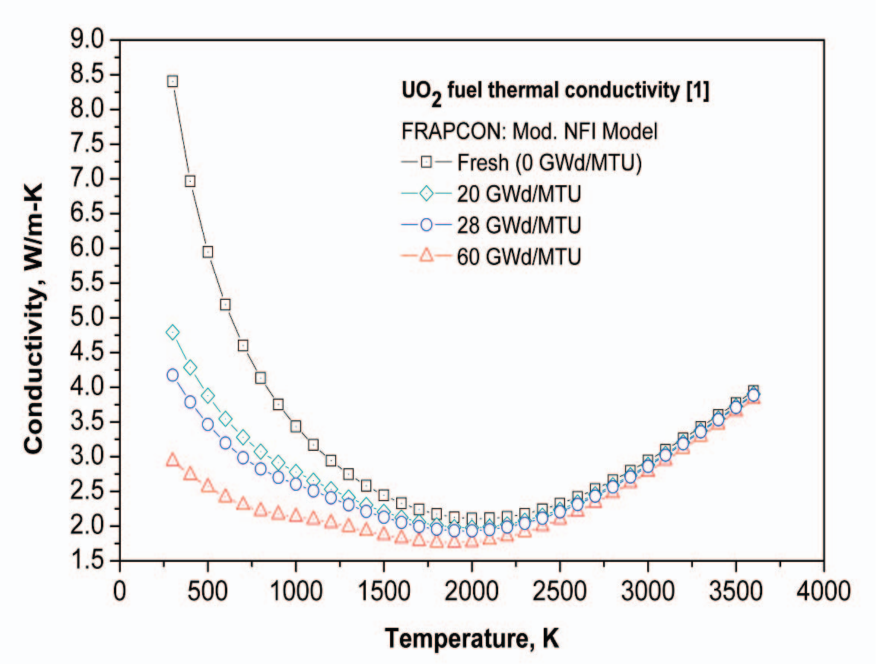
Findings

- Criteria to judge $PCT_{BLOWDOWN}$ or $PCT_{REFLOOD}$
  - 2 or 3 peak points of the cladding temperature responses
  - Unclear distinction of PCT's at blowdown or reflood
  - Criteria: core water level recovered to core bottom.
- Uncertainty in Blowdown quenching
  - In SKN Units 3&4, $PCT_{REFLOOD} << PCT_{BLOWDOWN}$
  - Due to the falling of the large water in upper head and flow path through the guide tubes,
  - To confirm the validity and uncertainty on blowdown quenching, further study needed.
- Improvement of reflood model of MARS-KS code

4. Results and Findings

Further Issues

- Degradation of thermal conductivity of pellet with burnup
  - Effect of Burnup considered for other fuel parameters ?
  - How to consider it for uncertainty evaluation ?
- Long term core cooling
  - Potential to Core Blockage by debris and others
- Application to CANDU
  - Coupling with Neutronics to consider void power pulse
  - Determination of Number of Failed Rods



OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations

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

Coupled Code Analysis of the Rod Withdrawal at Power Accident including Uncertainty Evaluation Using CIAU-TN Method

Nikola Čavlina, Davor Grgić, FER - University of Zagreb
Alessandro Petruzzi, GRNSPG - University of Pisa
Tomislav Bajs, ENCONET d.o.o.

Contents

- Background for the analysis
- Coupled code methodology and NPP Krško model
- Results of the Krško RWAP analysis
- CIAU-TN
- Results of the uncertainty analysis
- Conclusion

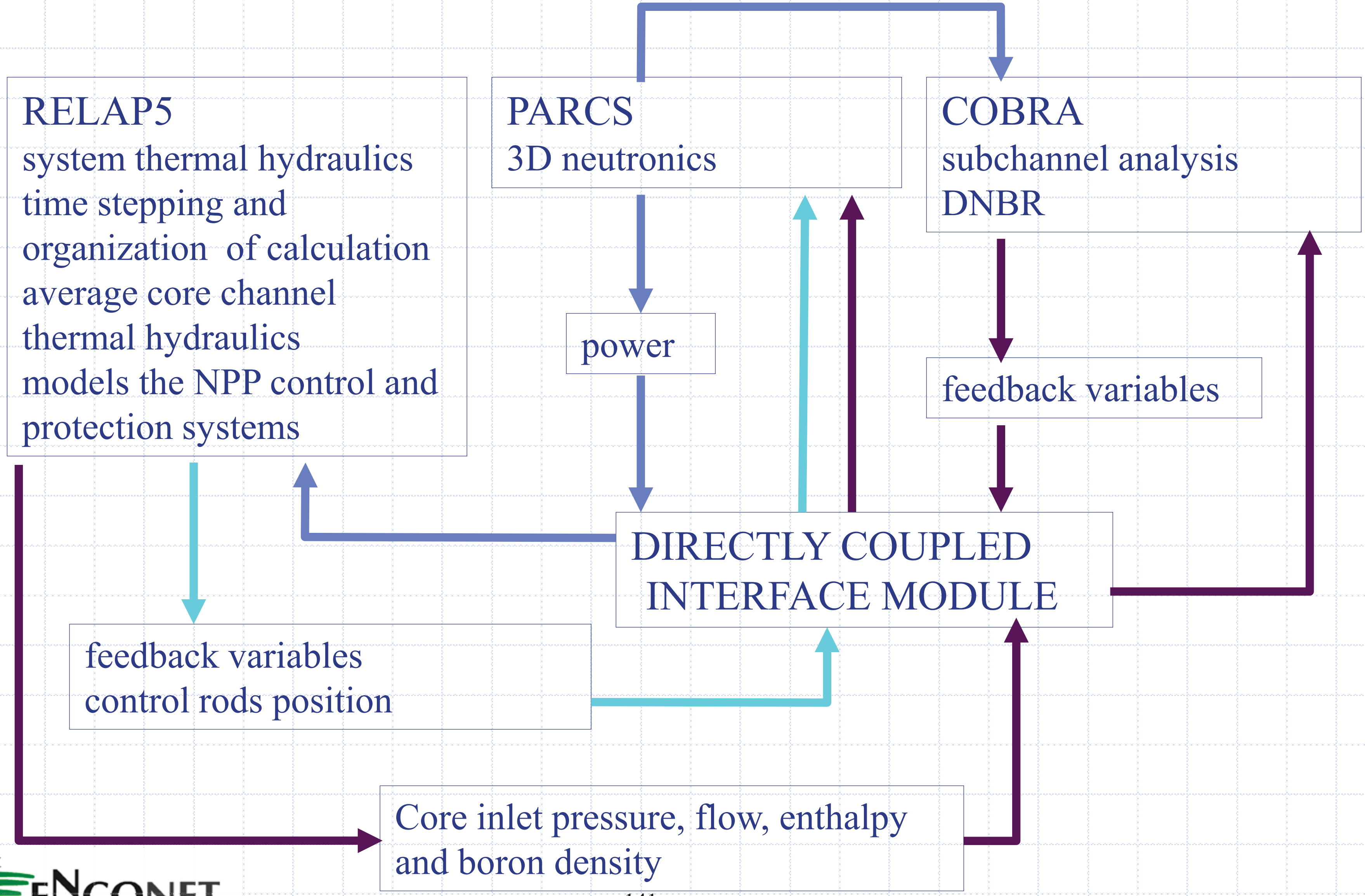
Background for the Analysis

- NPP Krško evaluates possible Resistance Temperature Detectors (RTD) bypass removal and use of thermowell for the average temperature measurement
- Fast acting thermowells are embedded as part of a primary loop pipe wall and their response time is slower
- Different coolant temperature measurement delays influences Overtemperature ΔT (OTΔT) and Overpower ΔT (OPΔT) protection
- Uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal at power can be DNBR or overpower limiting if protection system does not provde timely reactor scram

RWAP Phenomenology

- Reactivity insertion causes an increase in core power.
- Steam generator (SG) heat removal lags behind the core power generation rate
- SG pressure rises to the relief or safety valves setpoint
- Unbalanced heat removal rate causes the reactor coolant temperature rise until reactor trip occurs

Coupled Code Methodology



RELAP5
system thermal hydraulics
time stepping and
organization of calculation
average core channel
thermal hydraulics
models the NPP control and
protection systems

PARCS
3D neutronics

COBRA
subchannel analysis
DNBR

power

feedback variables

DIRECTLY COUPLED
INTERFACE MODULE

feedback variables
control rods position

Core inlet pressure, flow, enthalpy
and boron density

Calculation Procedure

- Preparation of burnup and thermal hydraulics dependent neutron cross sections for material compositions present in the core,
- 3D depletion calculation including preparation of reactivity data for RELAP5 point kinetics and 3D burnup and history variables distributions for coupled code
- Point kinetics calculation to explore main influence of changes in reactor coolant temperature measurements and time delays
- Coupled code calculation for different RWAP withdrawal rates
- Hot channel DNBR calculation for selected state points

Coupled NPP Krško Model

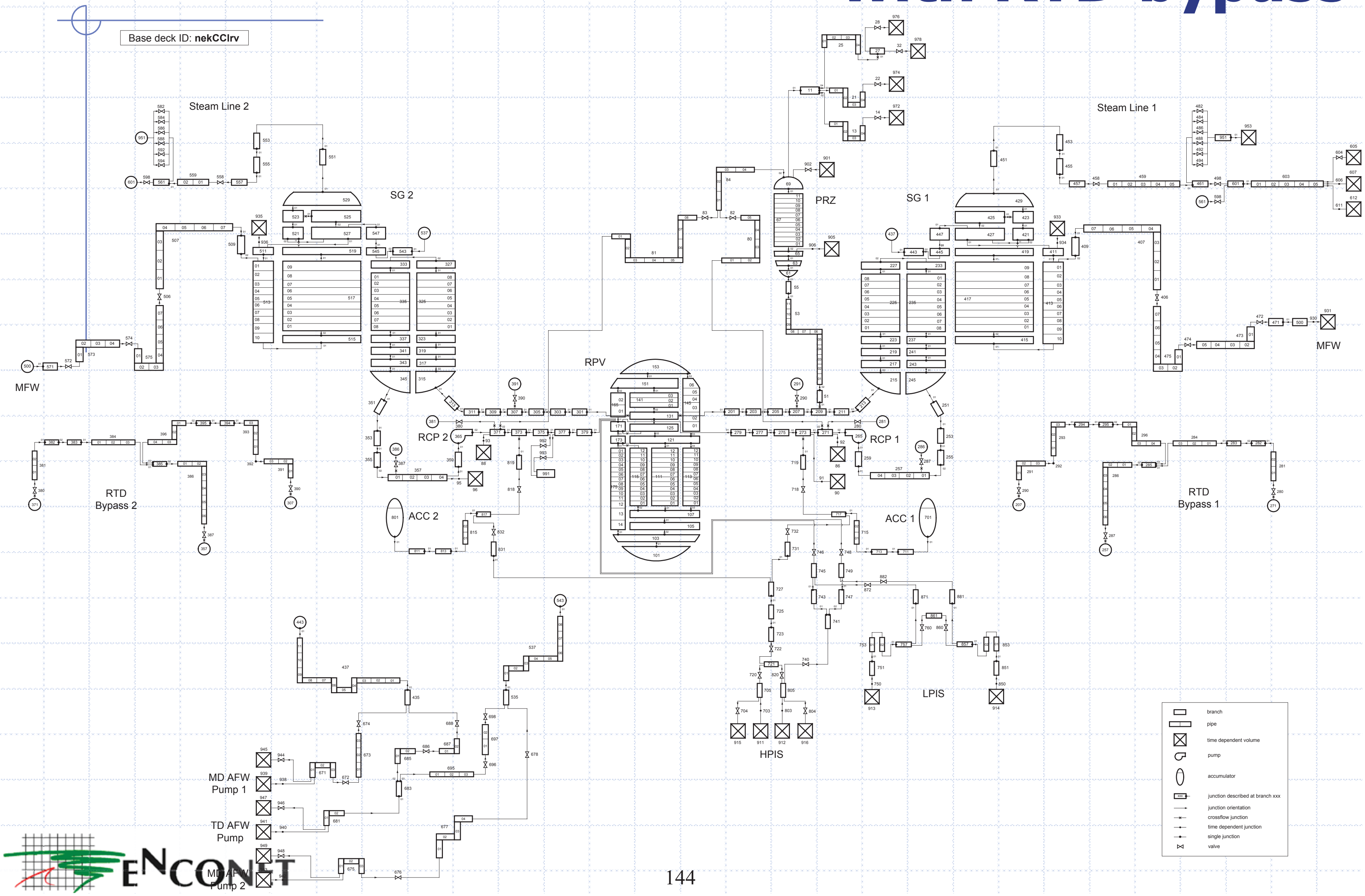
ITEM	VALUE
Heat structures	995
Mesh points	9656
Hydraulic nodes	909
Junctions	964

ITEM	VALUE
Hydraulic stacks in core	24
Core active regions	18
Mesh points in core	6912
Hydraulic nodes in core	432
Junctions in core	306
Heat structures in each SG	58
Hydraulic nodes in each SG	38
Junctions in SG (each)	41

TH model core

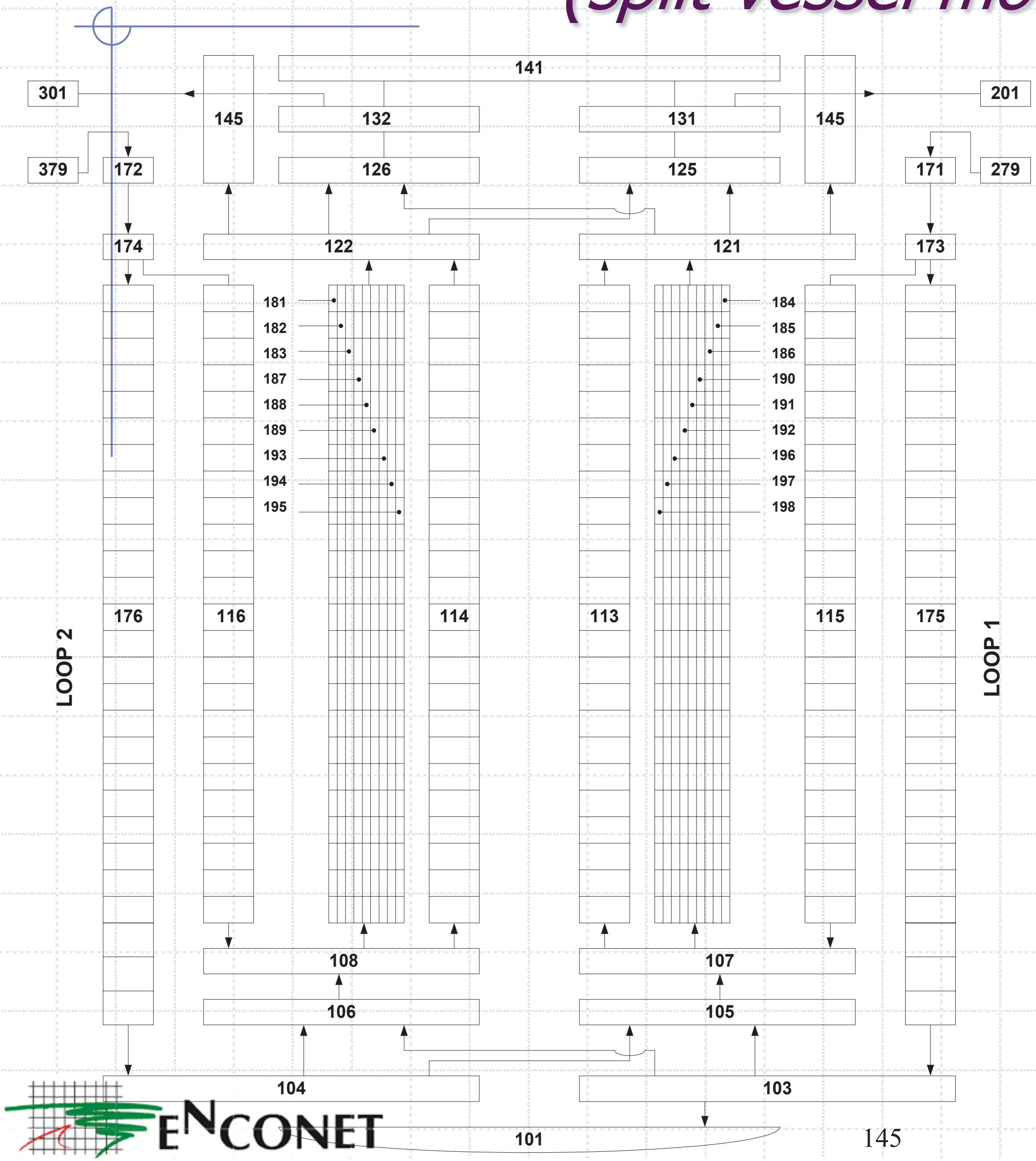
- R5PA: 18+2 pipes, 24 axial subdivisions, 18 heat structures (16/10 mesh points)
- Total 1007 control volumes, 1060 junctions, 1140 heat structures (10095 mesh points)
- Core mixing according to the W supplied data

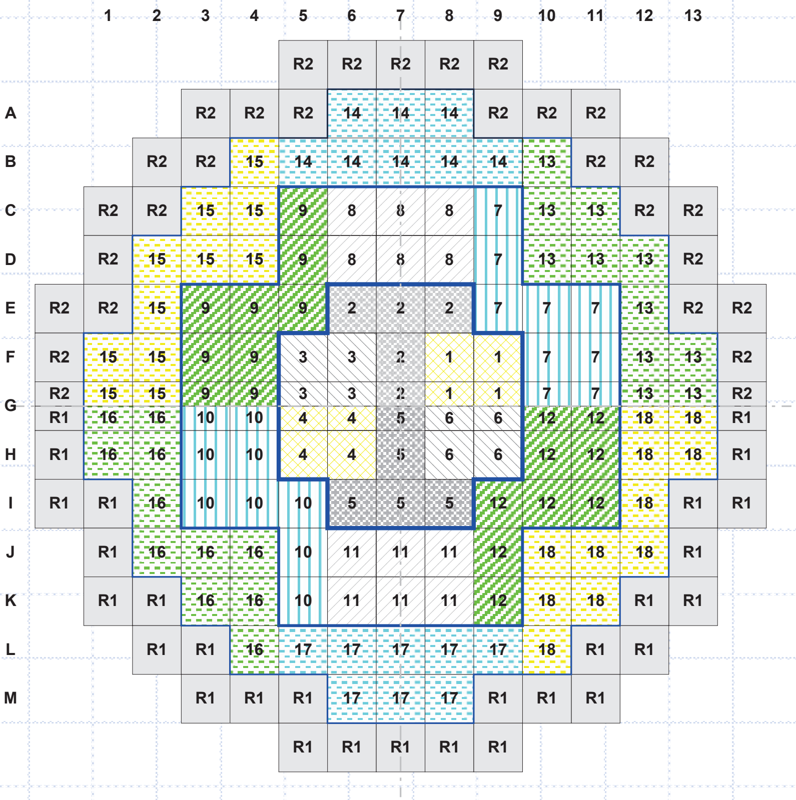
NPP Krško RELAP5 nodalization with RTD bypass



NPP Krško Model

(split vessel model)





		1	2	3	4	5	6	7	8	9	10	11	12	13	
						R2	R2	R2	R2	R2					
A				R2	R2	R2	14	14	14	R2	R2	R2			
B			R2	R2	15	14	14	14	14	14	13	R2	R2		
C		R2	R2	15	15	9	8	8	8	7	13	13	R2	R2	
D		R2	15	15	15	9	8	8	8	7	13	13	13	R2	
E	R2	R2	15	9	9	9	2	2	2	7	7	7	13	R2	R2
F	R2	15	15	9	9	3	3	2	1	1	7	7	13	13	R2
G	R2	15	15	9	9	3	3	2	1	1	7	7	13	13	R2
	R1	16	16	10	10	4	4	5	6	6	12	12	18	18	R1
H	R1	16	16	10	10	4	4	5	6	6	12	12	18	18	R1
I	R1	R1	16	10	10	10	5	5	5	12	12	12	18	R1	R1
J		R1	16	16	16	10	11	11	11	12	18	18	18	R1	
K		R1	R1	16	16	10	11	11	11	12	18	18	R1	R1	
L			R1	R1	16	17	17	17	17	17	18	R1	R1		
M				R1	R1	R1	17	17	17	R1	R1	R1			
						R1	R1	R1	R1	R1					

Results of the Krško RWAP analysis

- RELAP5 analyses were performed to make an assessment of the influence of RCS temperature measurement delay for various reactivity insertion rates on physical parameters affecting the DNBR (e.g., core heat power, RCS average temperature and ΔT, fuel cladding temperature).
- Sensitivity analyses for different RCS temperature measurement delays and reactivity insertion rates

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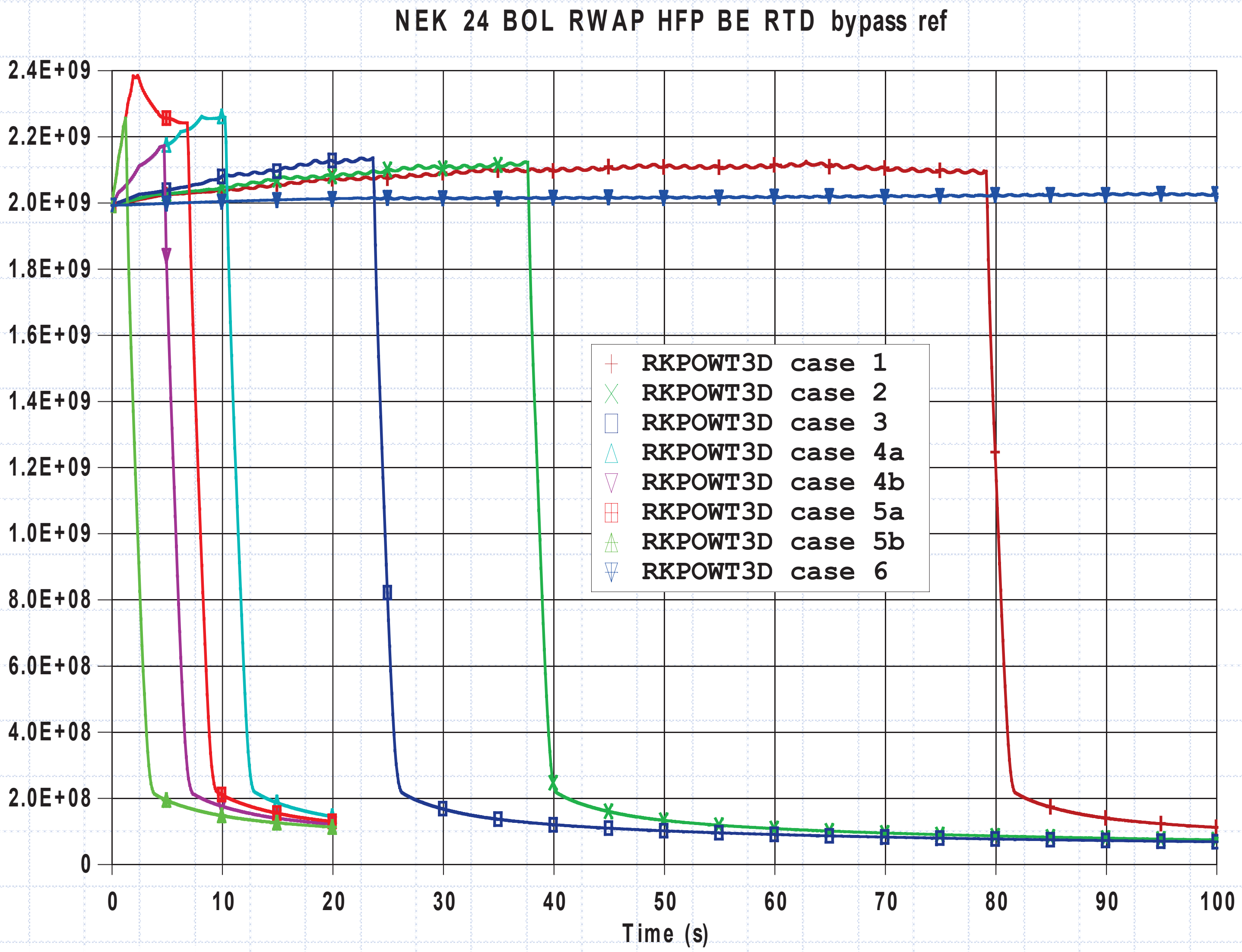
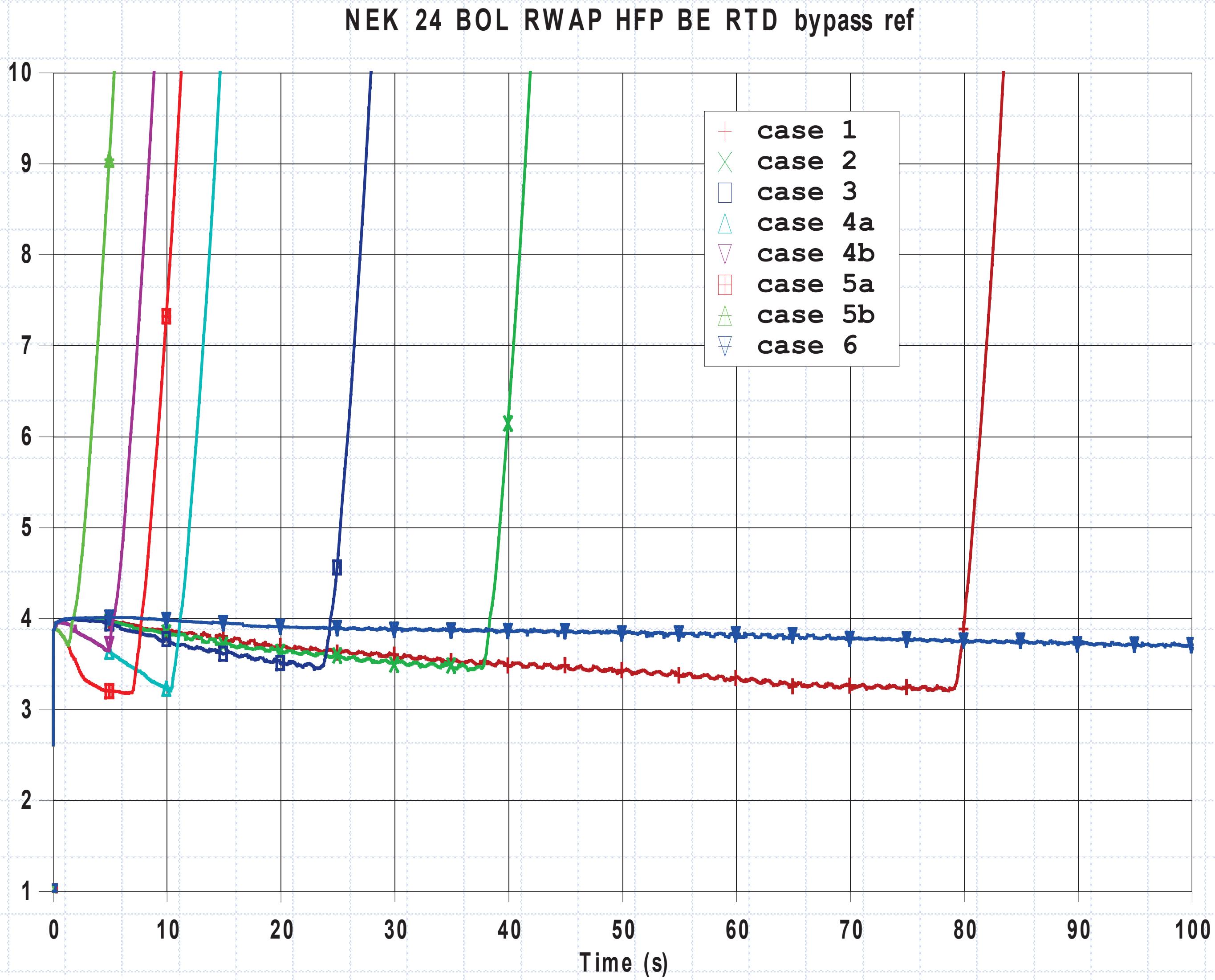
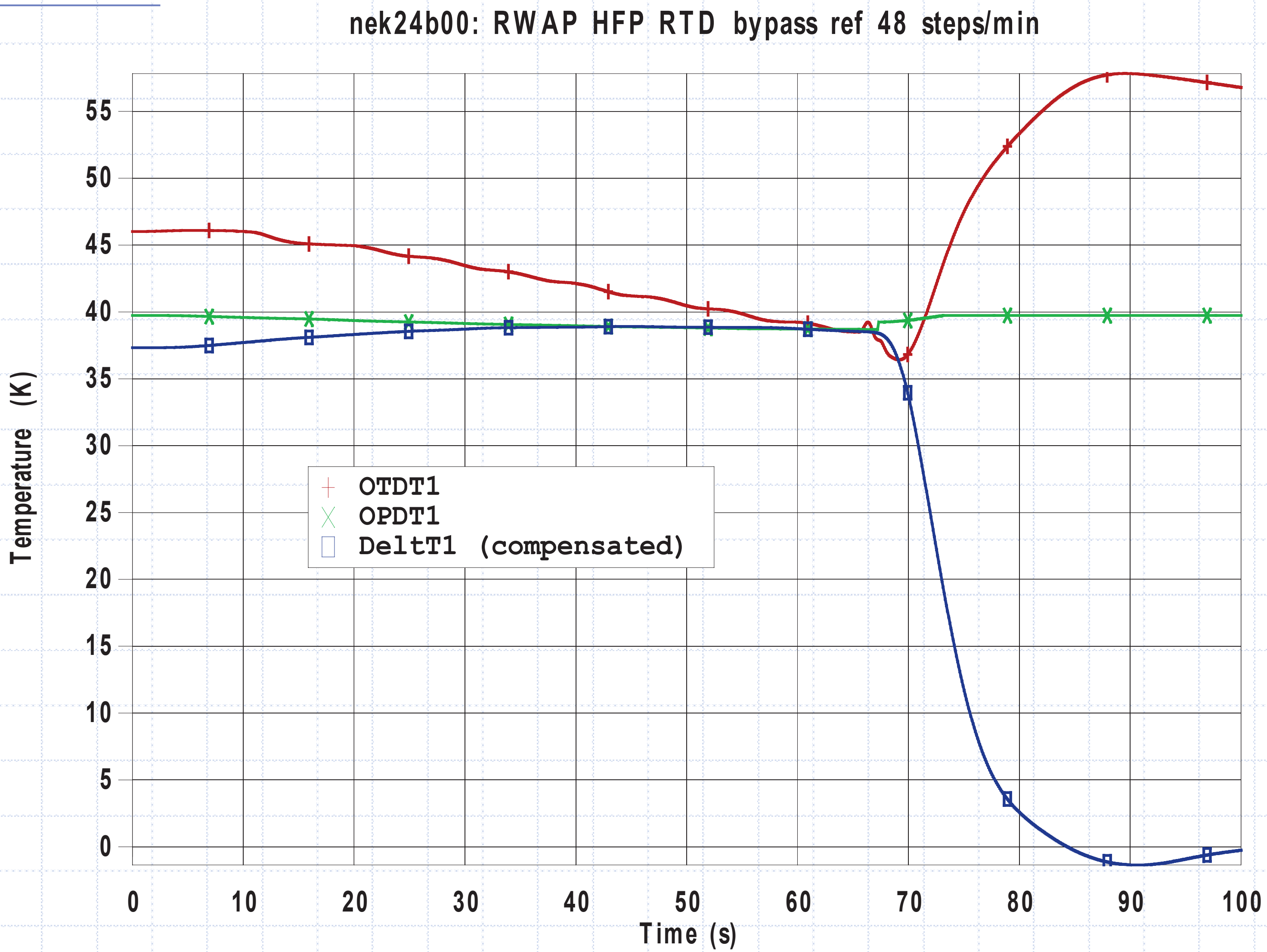
Nuclear power for different bank withdrawal rates



Coupled code - minimum DNBR for different bank withdrawal rates



Reactor protection setpoints



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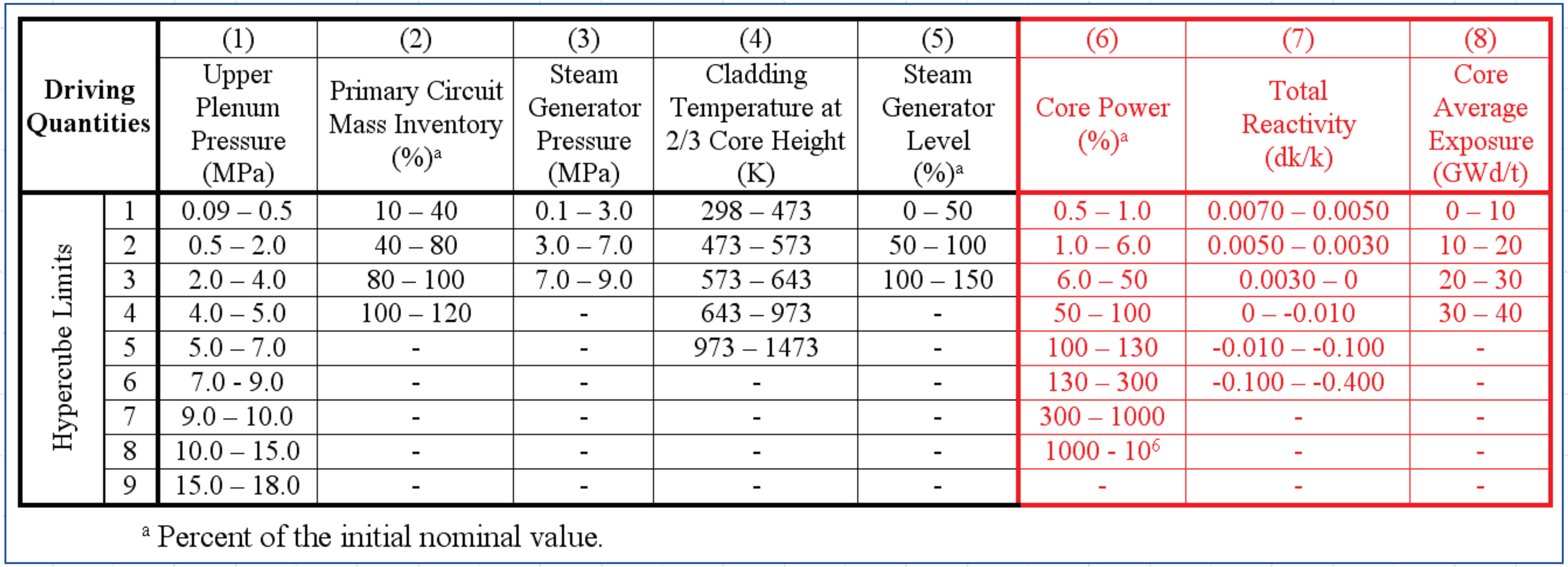
CIAU for 3D TH/NK Coupled Codes

- Extension of the ciau uncertainty analysis to the (3d) core power → ciau-tn
- Ciau-tn is based upon the same approach of ciau for thermal-hydraulics codes
- total reactivity and core average exposure are 'new' driving quantities:

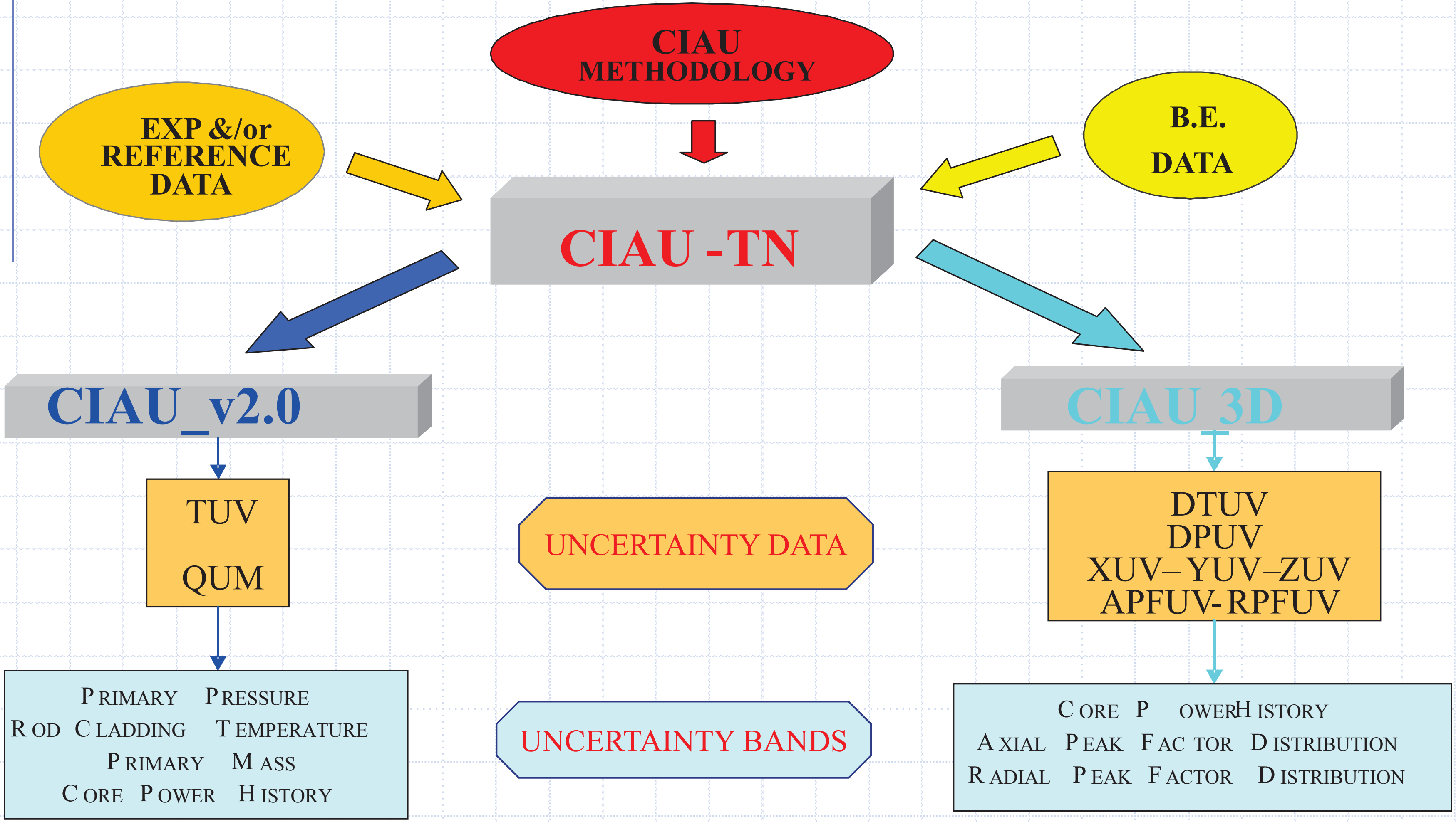


Driving Quantities		(1) Upper Plenum Pressure (MPa)	(2) Primary Circuit Mass Inventory (%)[a]	(3) Steam Generator Pressure (MPa)	(4) Cladding Temperature at 2/3 Core Height (K)	(5) Steam Generator Level (%)[a]	(6) Core Power (%)[a]	(7) Total Reactivity (dk/k)	(8) Core Average Exposure (GWd/t)
Hypercube Limits	1	0.09 – 0.5	10 – 40	0.1 – 3.0	298 – 473	0 – 50	0.5 – 1.0	0.0070 – 0.0050	0 – 10
	2	0.5 – 2.0	40 – 80	3.0 – 7.0	473 – 573	50 – 100	1.0 – 6.0	0.0050 – 0.0030	10 – 20
	3	2.0 – 4.0	80 – 100	7.0 – 9.0	573 – 643	100 – 150	6.0 – 50	0.0030 – 0	20 – 30
	4	4.0 – 5.0	100 – 120	-	643 – 973	-	50 – 100	0 – -0.010	30 – 40
	5	5.0 – 7.0	-	-	973 – 1473	-	100 – 130	-0.010 – -0.100	-
	6	7.0 - 9.0	-	-	-	-	130 – 300	-0.100 – -0.400	-
	7	9.0 – 10.0	-	-	-	-	300 – 1000	-	-
	8	10.0 – 15.0	-	-	-	-	1000 - 10^6	-	-
	9	15.0 – 18.0	-	-	-	-	-	-	-

[a] Percent of the initial nominal value.

CIAU for 3D TH/NK Coupled Codes

CIAU-TN: Simplified Diagram



CIAU METHODOLOGY

EXP &/or REFERENCE DATA

B.E. DATA

CIAU -TN

CIAU_v2.0

CIAU 3D

TUV

QUM

UNCERTAINTY DATA

DTUV
DPUV
XUV–YUV–ZUV
APFUV-RPFUV

PRIMARY PRESSURE
ROD CLADDING TEMPERATURE
PRIMARY MASS
CORE POWER HISTORY

UNCERTAINTY BANDS

CORE POWER HISTORY
AXIAL PEAK FACTOR DISTRIBUTION
RADIAL PEAK FACTOR DISTRIBUTION

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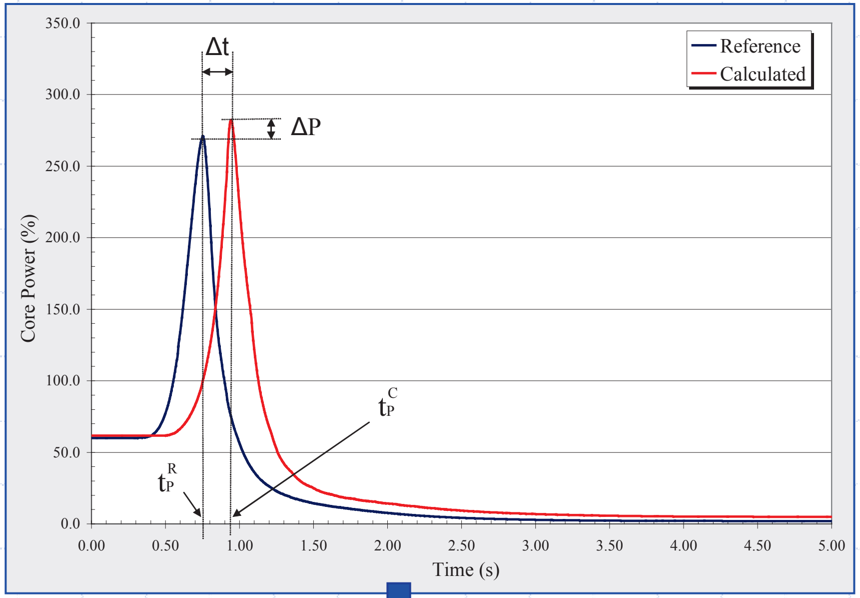
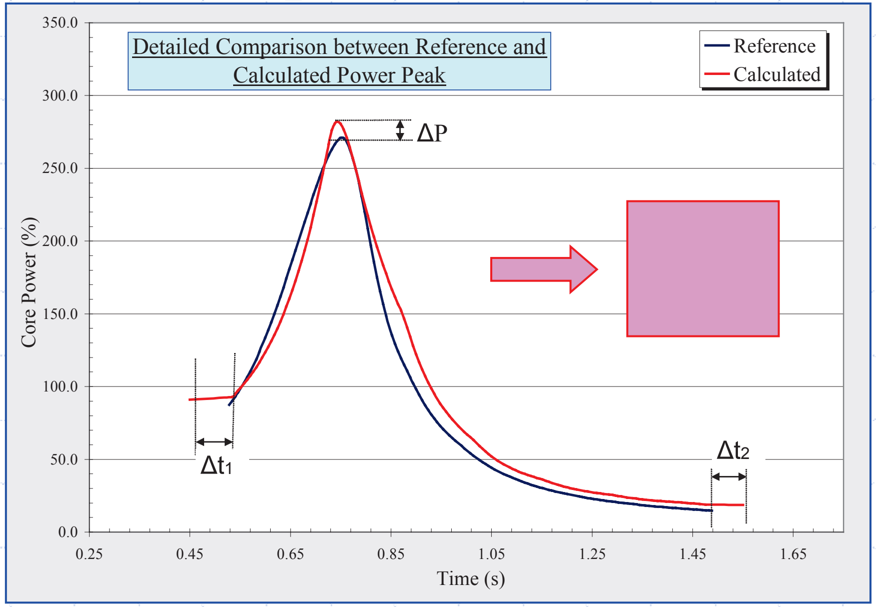
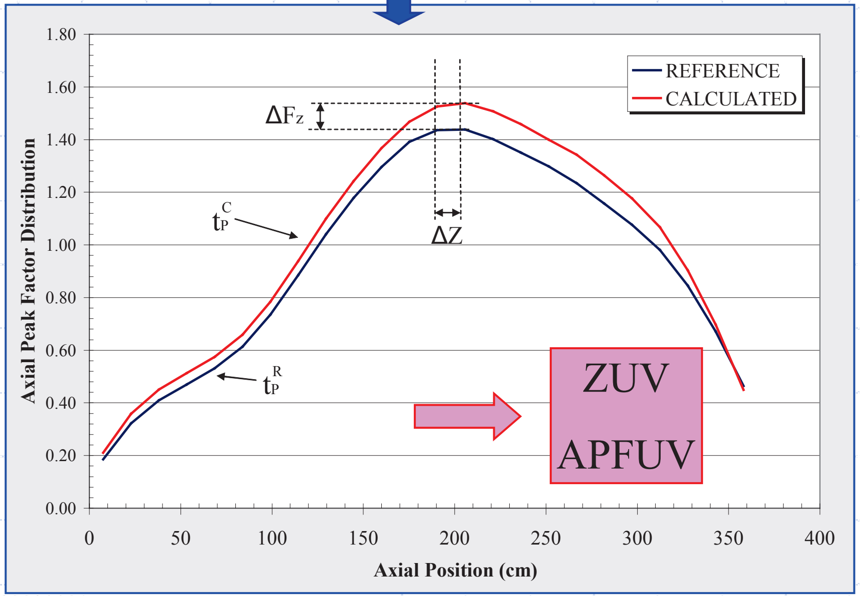
New uncertainty matrixes and vectors in CIAU-TN

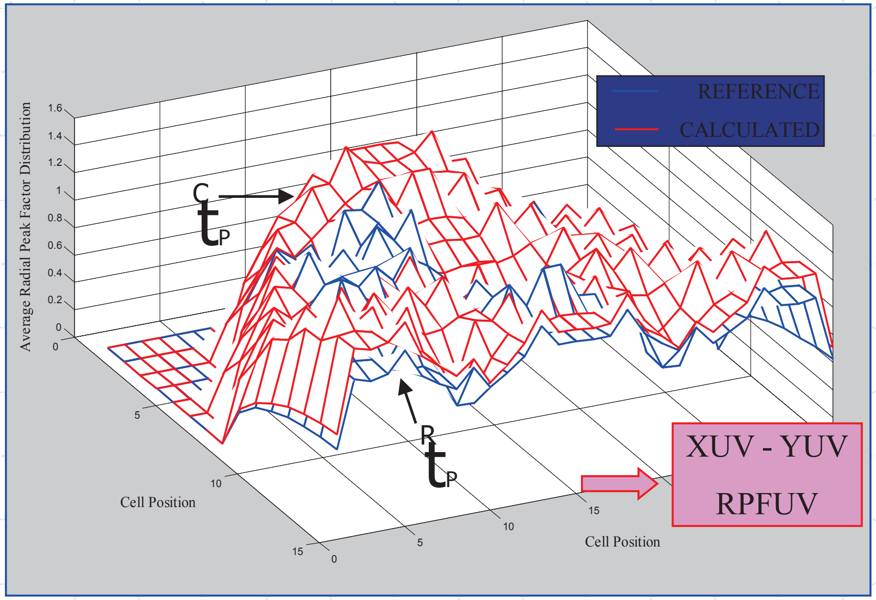
ID	DESCRIPTION	OBJECTIVE
DTUV	Detailed Time Uncertainty Vector	Uncertainty on the Time when the occurs
DPUV	Detailed Power Uncertainty Vector	Uncertainty on the Quantity of the Core Power
APFUV	Axial Peaking Factors (Fz) Uncertainty Vector	Uncertainty on the Quantity of Fz
ZUV	Axial (Z) Position Uncertainty Vector	Uncertainty on Axial Position of the Maximum Value Of Fz
RPFUV	Radial Peaking Factors (FR) Uncertainty Vector	Uncertainty on the Quantity of FR
XUV	Radial (X) Position Uncertainty Vector	Uncertainty on Radial (X) Position of The Maximum Value of FR
YUV	Radial (Y) Position Uncertainty Vector	Uncertainty on Radial (Y) Position of the Maximum Value of FR

CIAU for 3D TH/NK Coupled Codes









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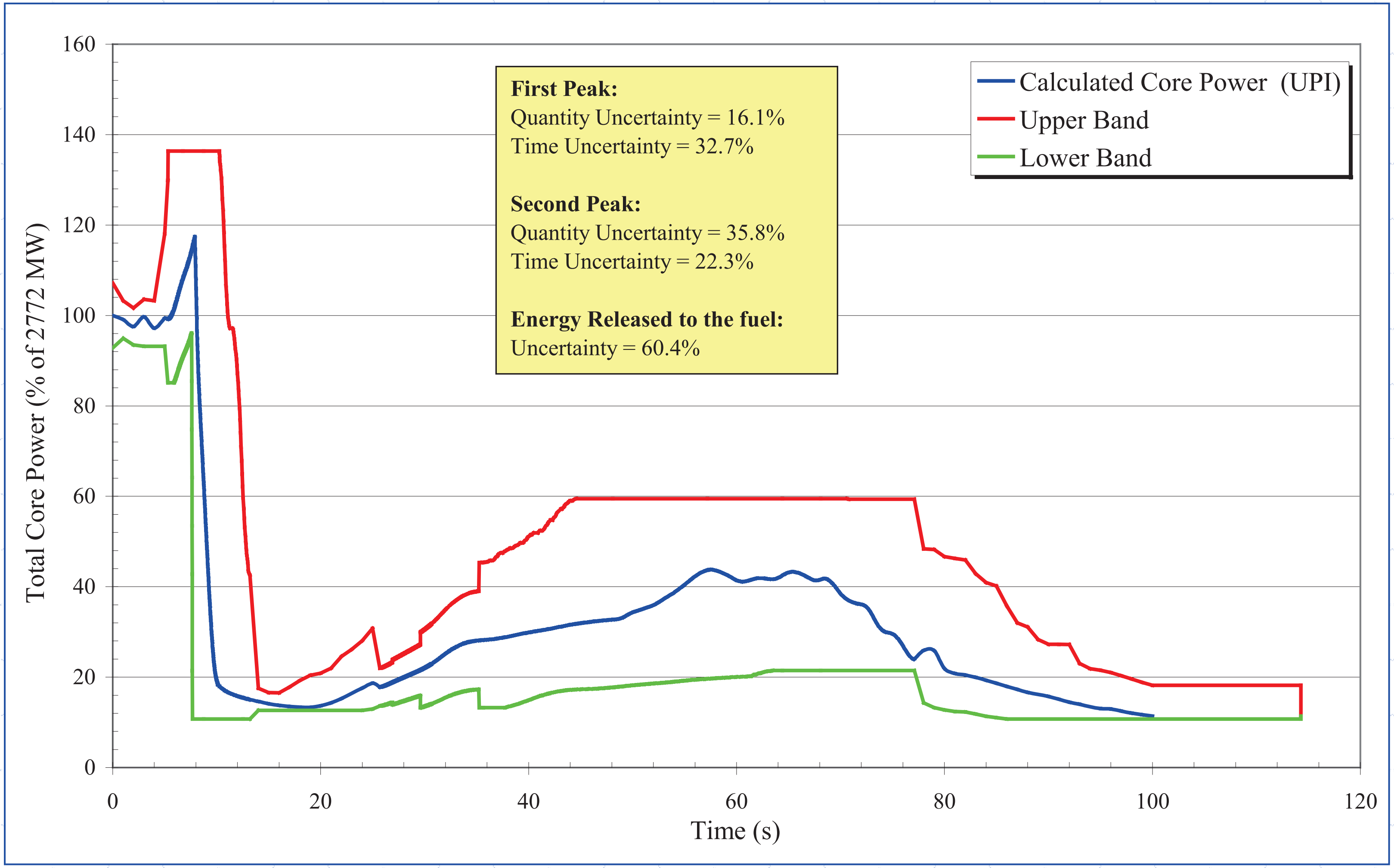
CIAU-TN: Application to MSLB

TOTAL CORE POWER



First Peak:
Quantity Uncertainty = 16.1%
Time Uncertainty = 32.7%

Second Peak:
Quantity Uncertainty = 35.8%
Time Uncertainty = 22.3%

Energy Released to the fuel:
Uncertainty = 60.4%

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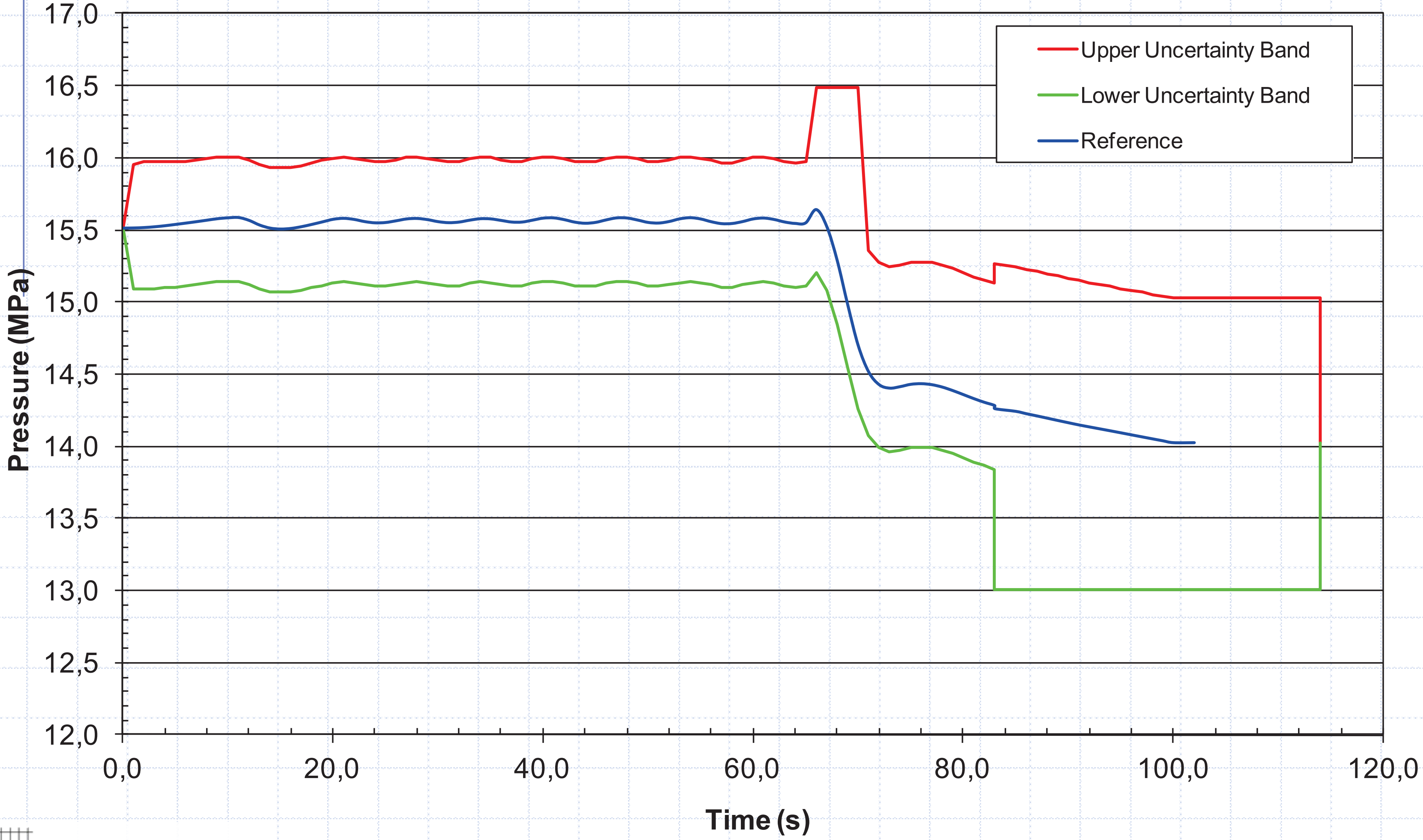
CIAU-TN: Application to RWAP

Primary pressure - derived uncertainty bands



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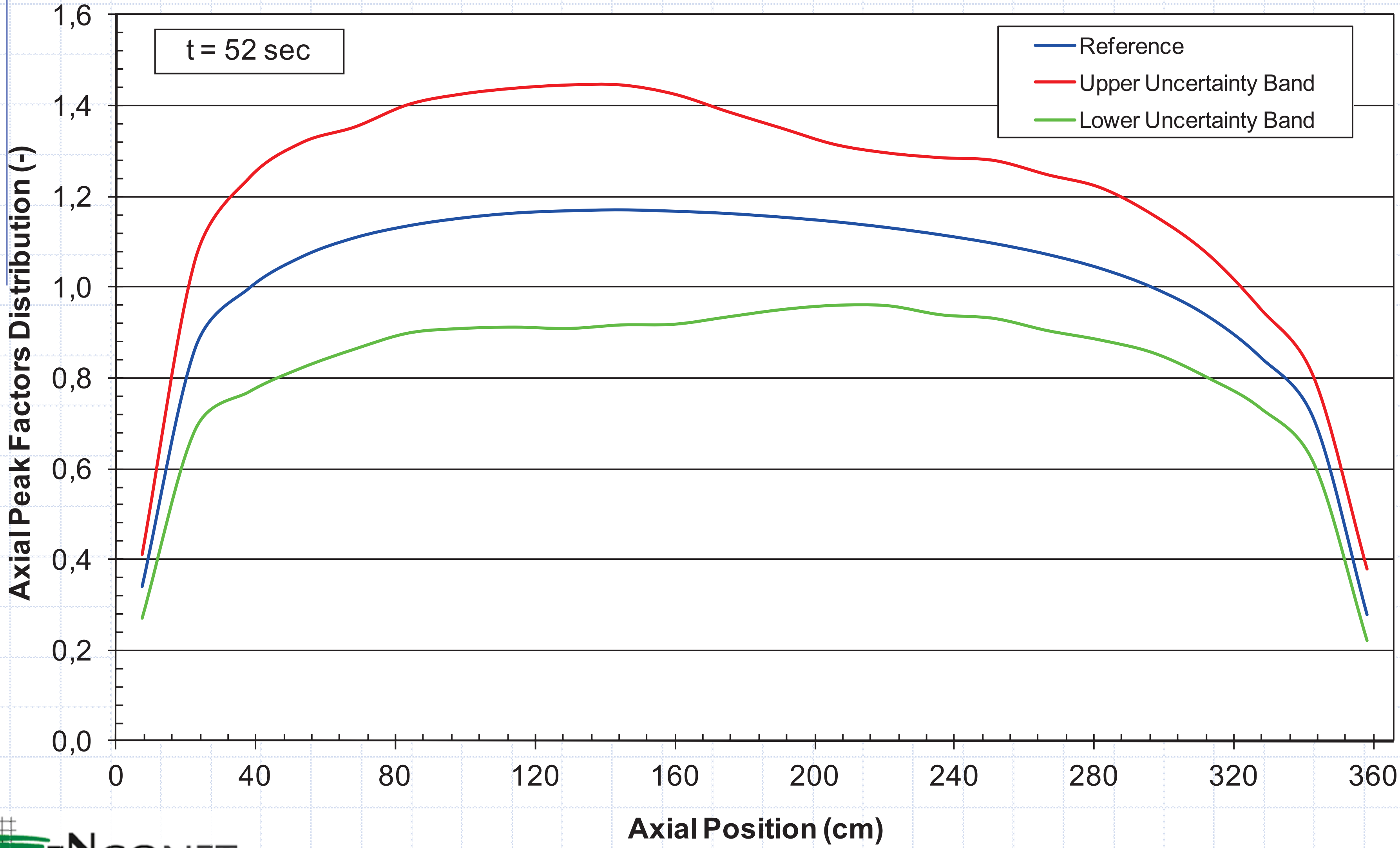
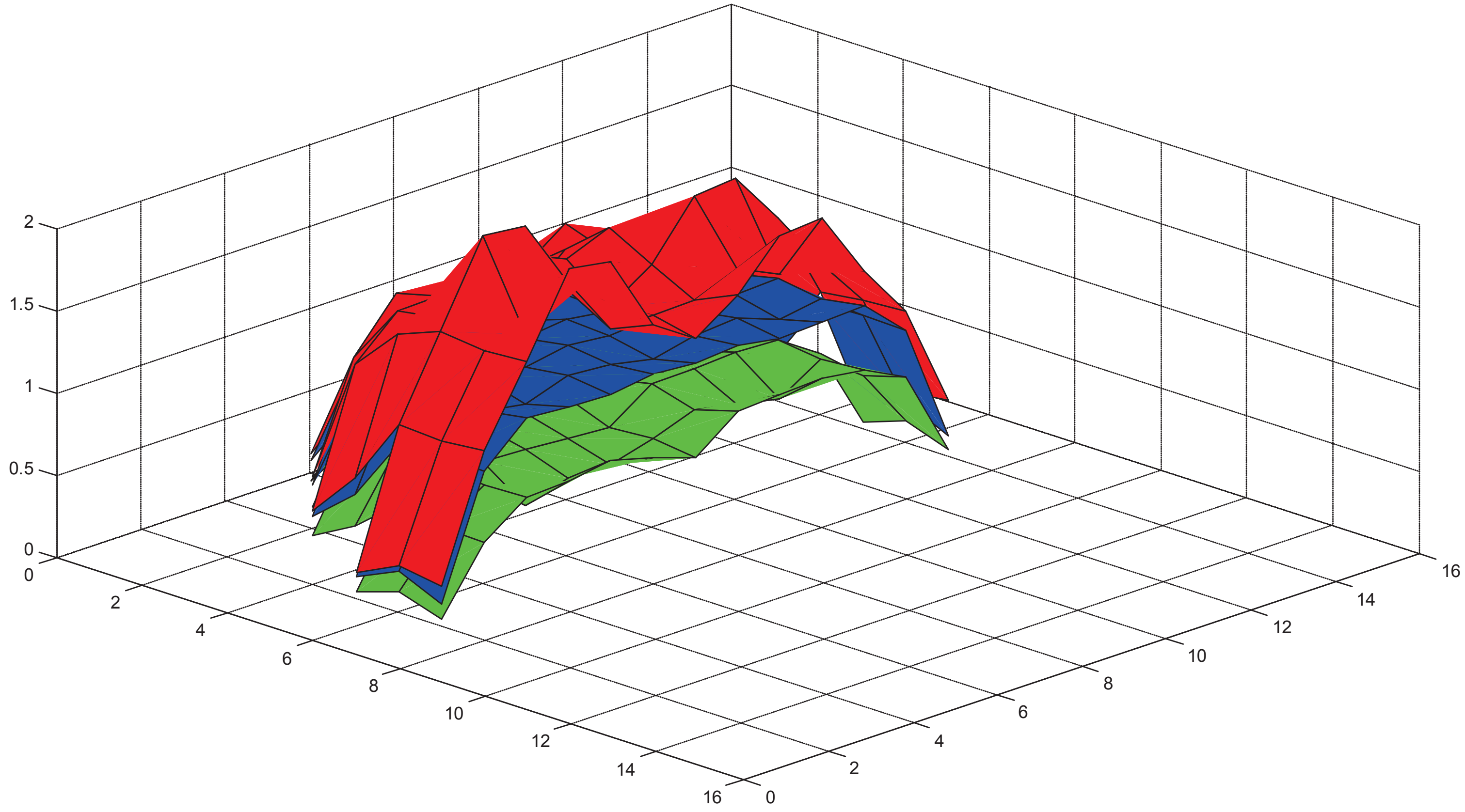
CIAU-TN: Application to RWAP

Axial pear power distribution at time of maximum power



CIAU-TN: Application to RWAP

Radial peak power distribution at time of maximum power - derived uncertainty bands



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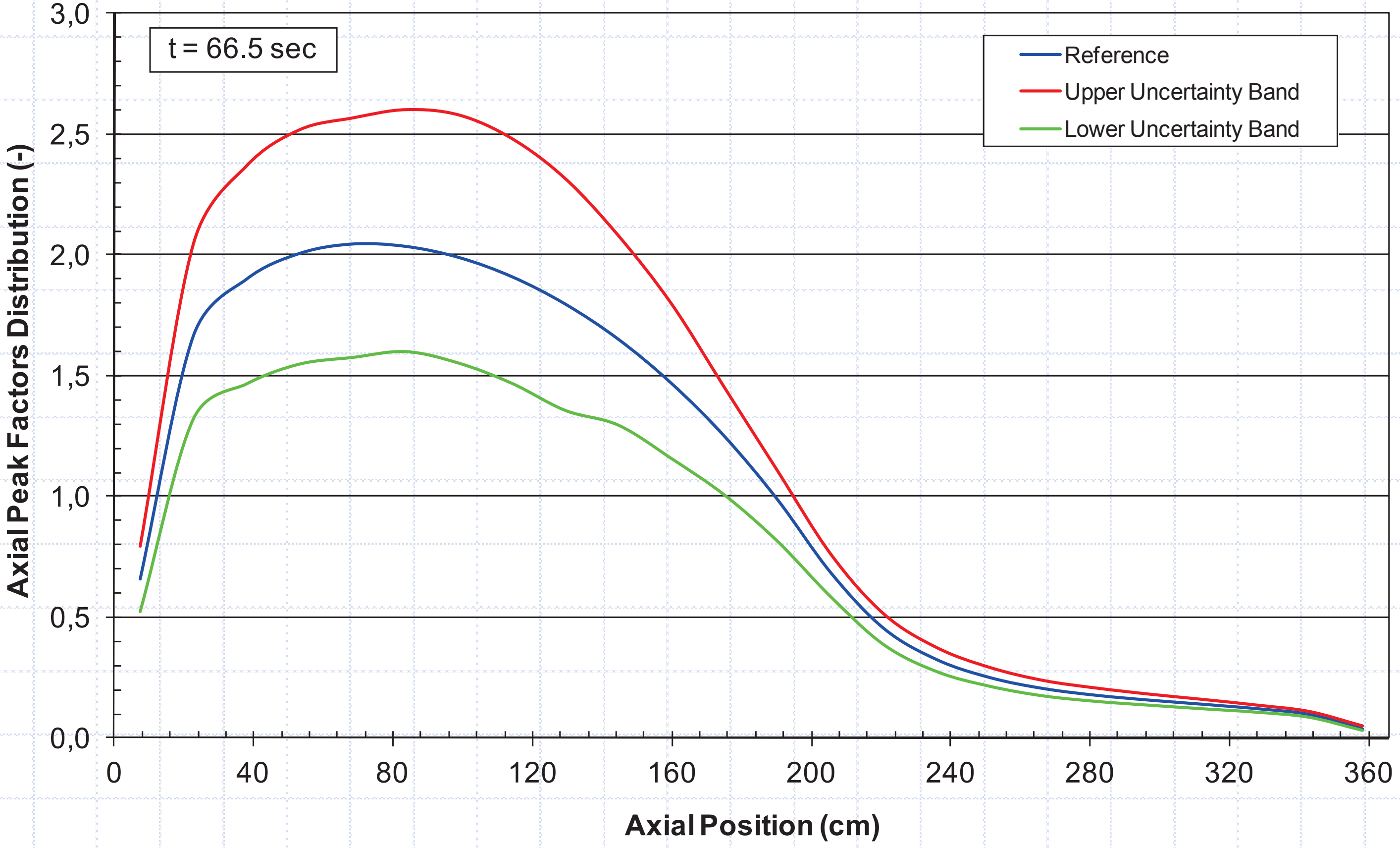
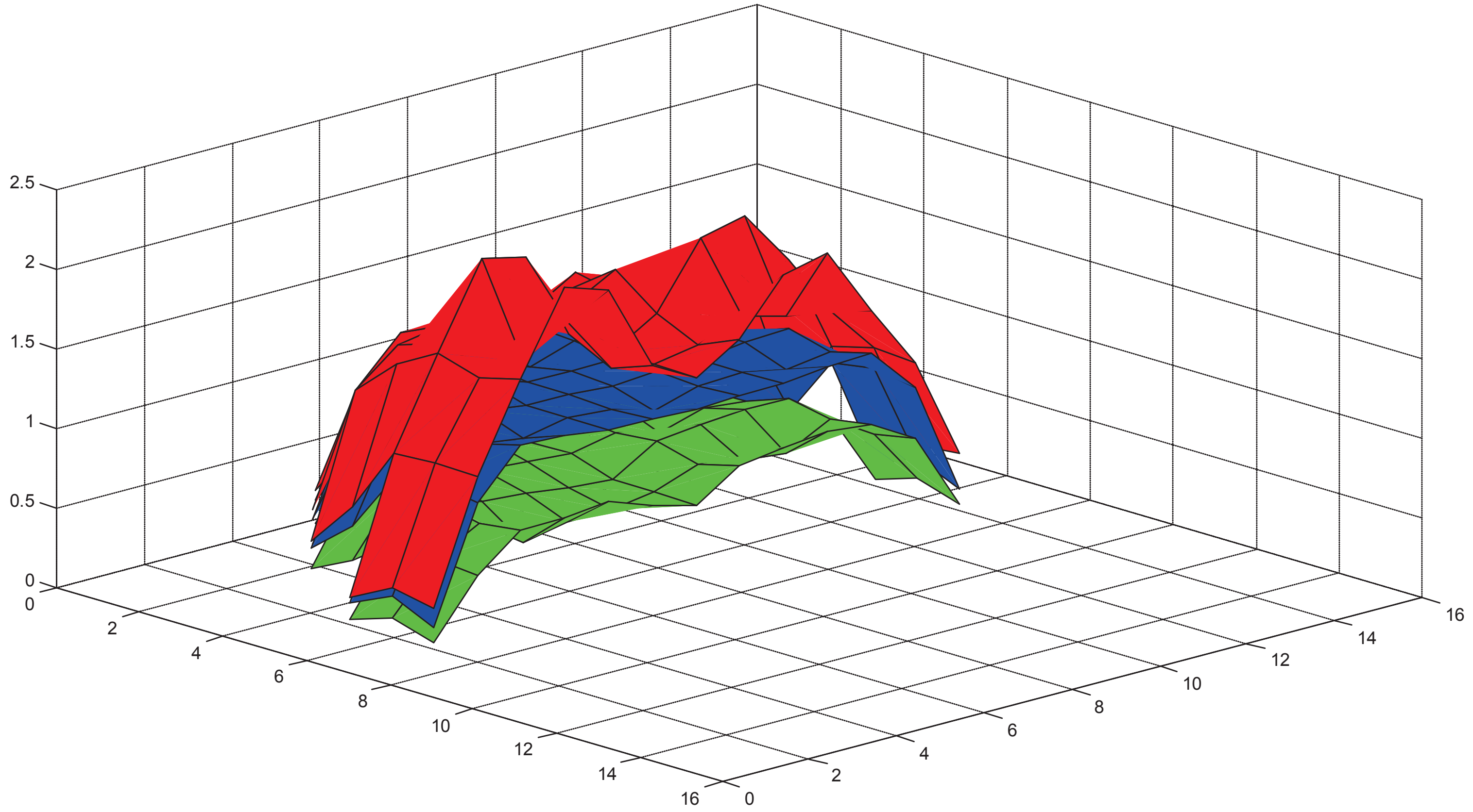
CIAU-TN: Application to RWAP

Axial pear power distribution at time of scram



CIAU-TN: Application to RWAP

Radial peak power distribution at time of scram



ENCONET

CIAU-TN Uncertainties for RWAP

- Core power: 23,7%
- Fz distribution:
  - time of maximum power: 24.6% ;
  - time of OTDT trip: 29.5%;
- Fxy distribution:
  - time of maximum power: 69.6%;
  - time of OTDT trip: 72.9%.

Conclusion

- Small influence of change in coolant temperature measurements and applicable delays was found in OPDT and OTDT protection setpoints
- Small increases in peaking factors are present for withdrawal from full power and corresponding increase of fuel temperature is mild for the reference case
- CIAU methodology minimizes the resources and the engineering judgements (=expertise) needed to perform the uncertainty evaluation

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Conclusion (continued)

- Largest uncertainties have been observed in the lower part of the core where highest peaking factors have been calculated.
- The results still indicate robustness to the localized effects.
- CIAU-TN hypercubes are not fully qualified for all core states so the discussed results are applicable only for qualitative discussion of uncertainty impact on the final outcome.

ENCONET

UNCERTAINTY ANALYSIS FOR LBLOCA OF WOLSUNG UNIT-I REACTOR USING MARS CODE

Byoung Sub Han
ENESYS Co.

Young Seok Bang
Korea Institute of Nuclear Safety

Contents

I. Introduction

II. System Representation

III. Uncertainty Parameter Selection

IV. Analysis

V. Conclusion & Remarks

I. Introduction

Wolsung Unit-I
MARS Code
Purpose of Study
Target of Analysis

Wolsung Unit-I

- At Present
  - 22 NPPs in commercial operation in Korea
    - 18 PWRs
      - 6 Westinghouse Model PWR
      - 2 Framatom PWR
      - 10 KSNP PWR
    - 4 PHWRs
      - 700 MWe class CANDU 6 Model
- Wolsung Unit-I
  - designed by Atomic Energy of Canada Limited (AECL)
  - went into service in 1983
  - got refurbishment process started in 2009
    - Pressure tube and calandria tube replacement
    - Some Design Improvements(Emergency Core Cooling System, Set-points reset etc)
    - Hope to operate for another 25 years

MARS Code

- Developed by KAERI (Korea Atomic Energy Research Institute)
- for a multi-dimensional and multi-purpose realistic thermal-hydraulic system analysis of light water reactor transients
- The backbones of MARS
  - RELAP5/MOD3.2 : System Analysis
  - COBRA-TF : 3-D Vessel Analysis
  - Basically developed for the simulation of PWR type vertical fuel channel models
  - Recently features of RELAP/CANDU like the horizontal core channels (CANCHAN) are developed
    - Used/verified in the analysis of safety of CANDU reactors

Purpose of Study

- To Increase Safety of Operation NPP
  - Develop MARS CANDU for the Evaluation of CANDU T/H
  - To Audit/Evaluate CANDU Safety
    - ✓T/H Model, Methodology Development
    - ✓Model Development of Core Neutron Dynamic(considering unique VCR)
  - Increase Credibility of MARS/CANDU

Target of Analysis

Target Plant

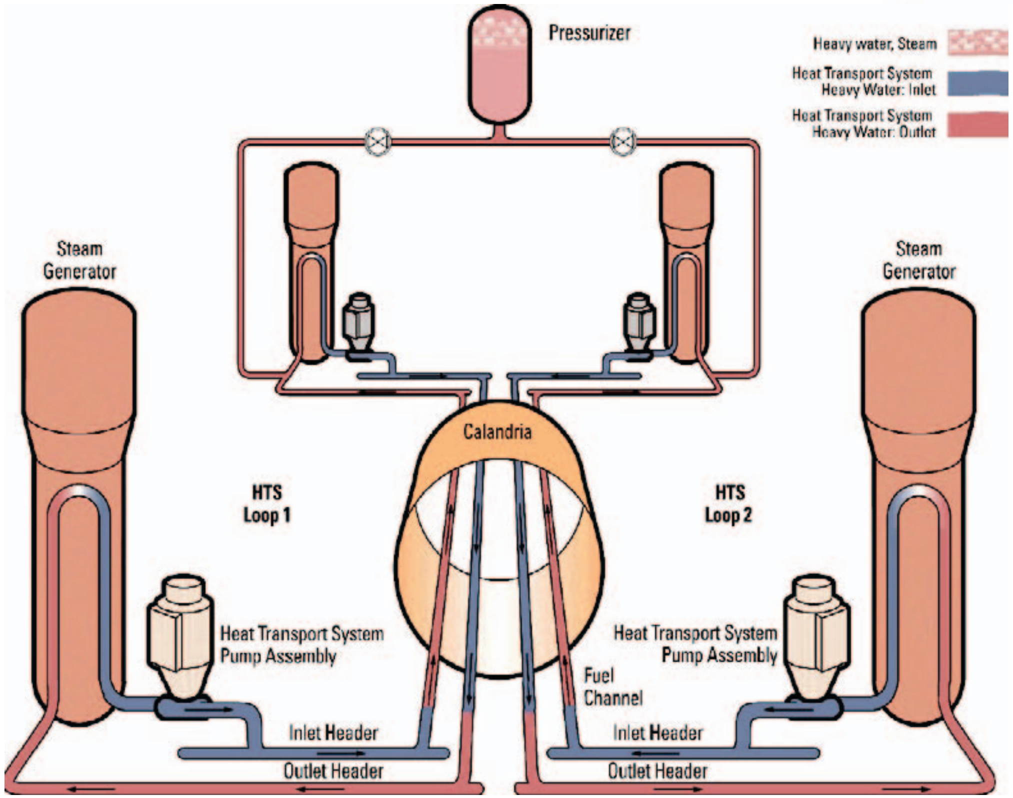
- Wolsung Unit-I(CANDU6)

Scenario

- 35% RiH LOCA

Acceptance Criteria

- PTC, Fuel Failure

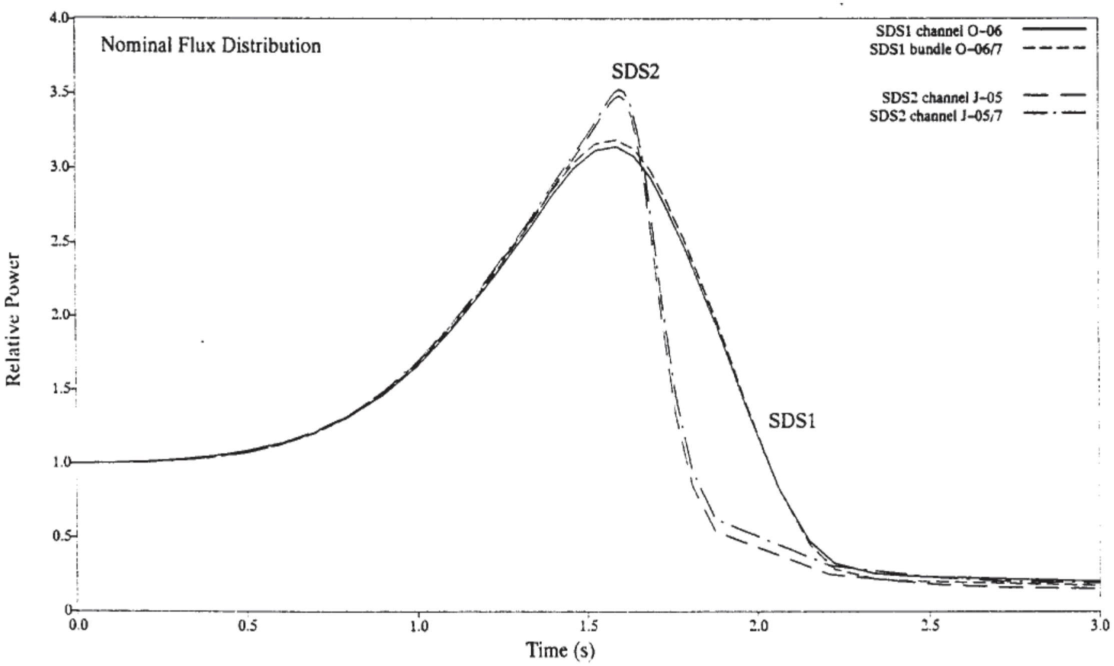


❖Target Phase of LOCA : Phase 1-2 (0~1000 sec)

- Phase 1 – Initial Power Pulse
- Phase 2 – Early Blowdown Cooling
- Phase 3 – Late Blowdown Cooling and ECC Refill
- Phase 4 – Long-Term Cooling

❖ Phase 1 – Initial Power Pulse

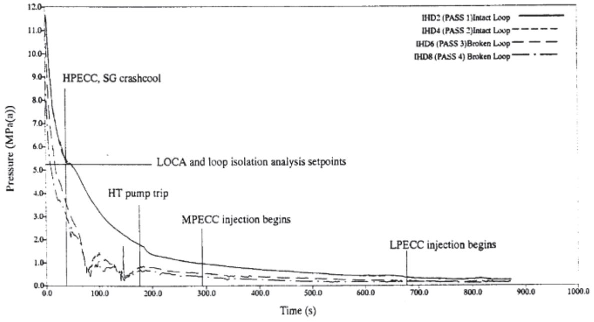
- Break Open
- Reactivity Increase by Core Void Formation
- Power Increase between SDS1, SDS2 Operation and Reactor Trip



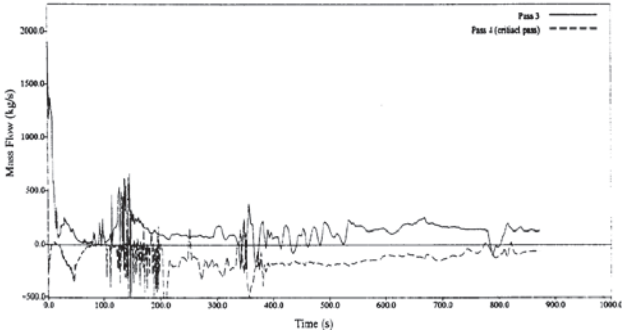
❖ Phase 2 – Early Blowdown Cooling

- Continueous blowdown
- Break Flow Decrease
- RIH Pressure Decrease & HP ECCS starts
- MSSV Open

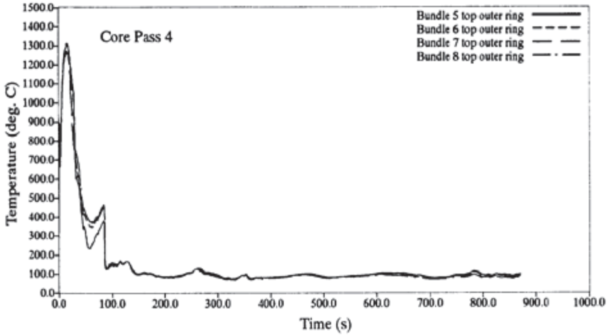
RIH Pressure



Pass Channel Flow



Critical Pass PCT



II. System Representation

Primary System

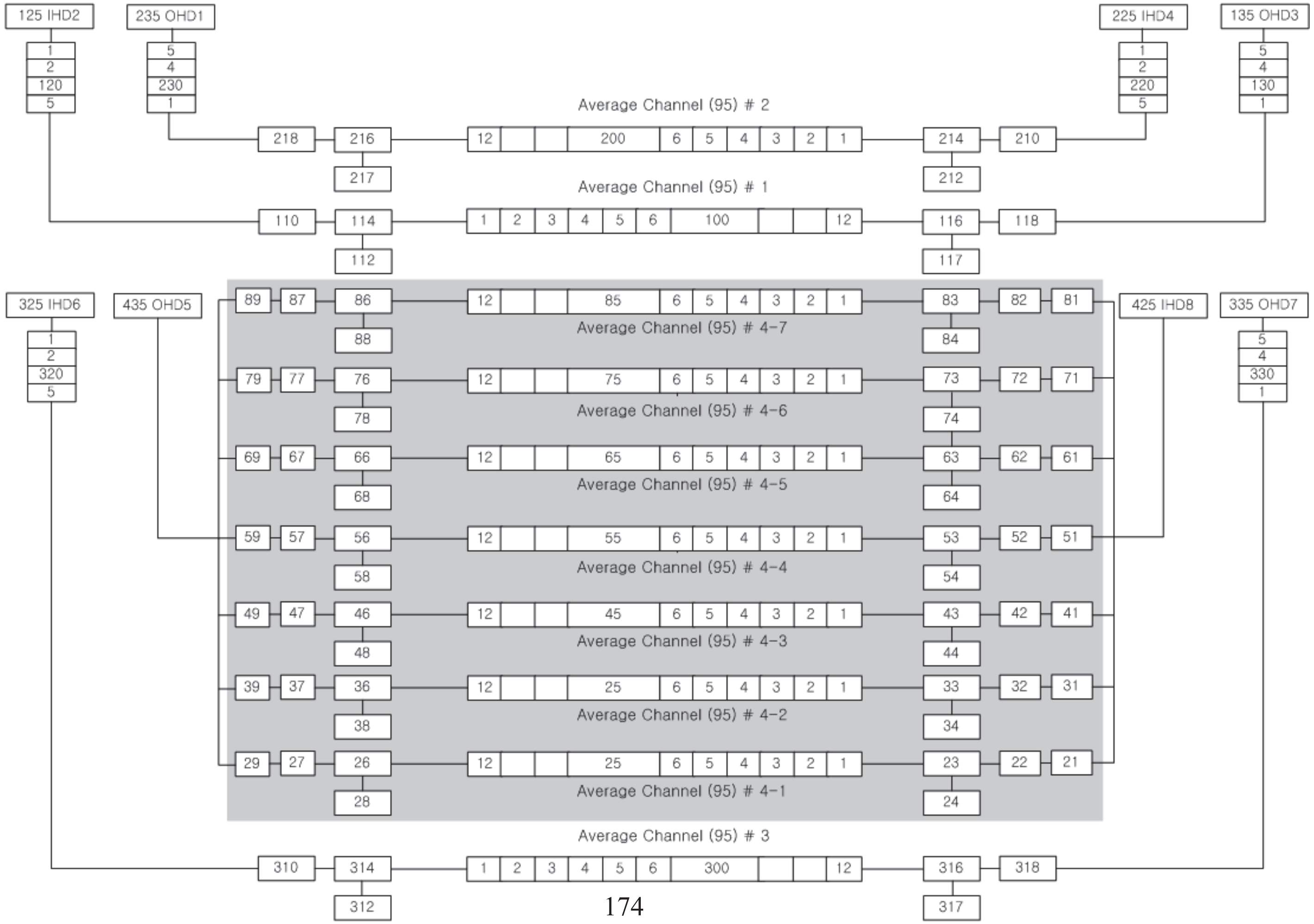
Fuel & Channel

Secondary System

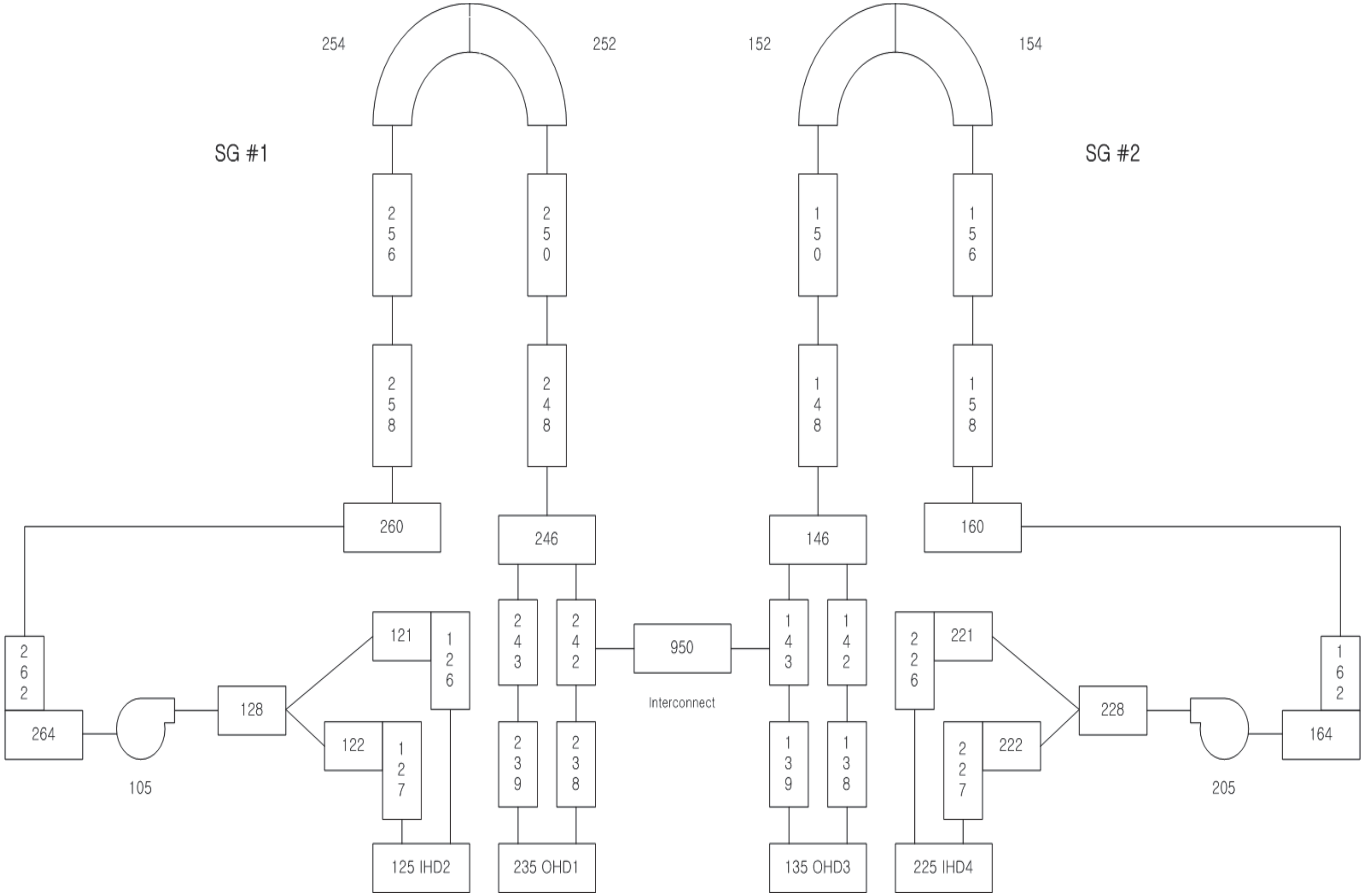
ECCS System

Primary System(1/2)

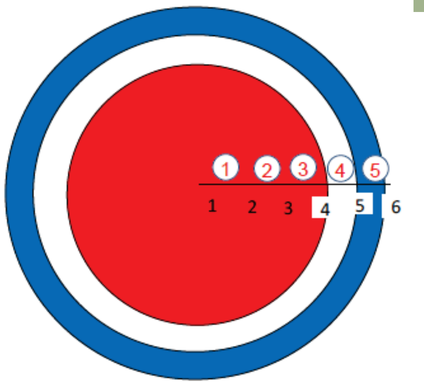
3 Quarter Core + 7 X 1/7 Interest Core Group



Primary System(2/2)

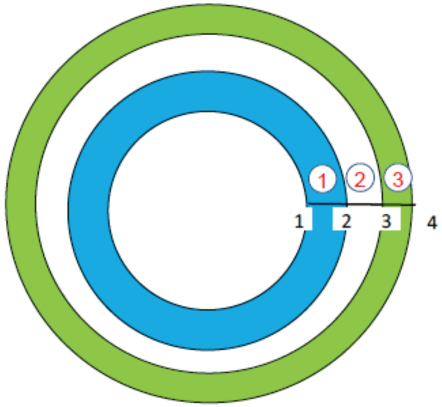


Fuel & Channel



Fuel Pin Heat Structure

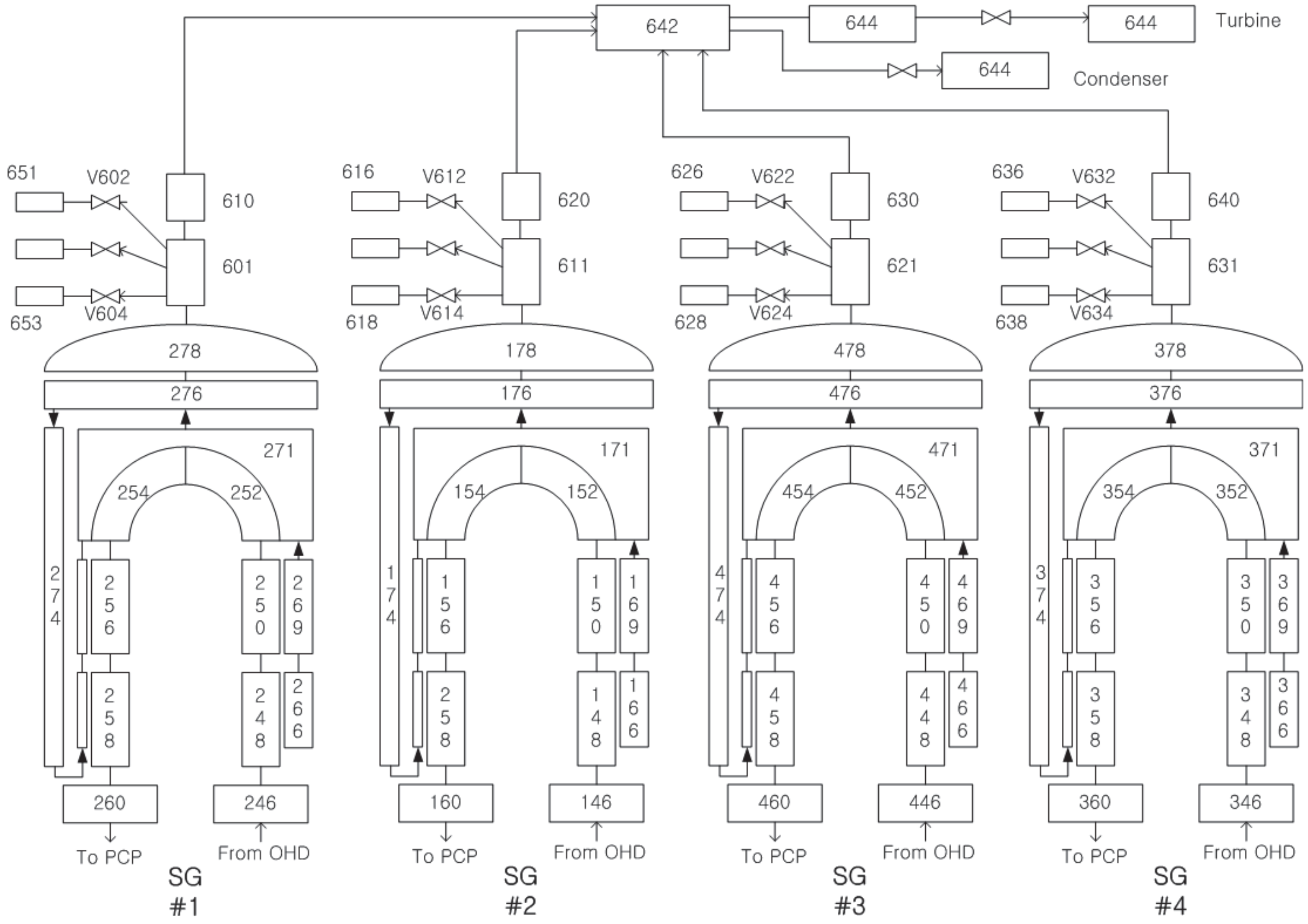
UO2-UO2-UO2-He-Zr



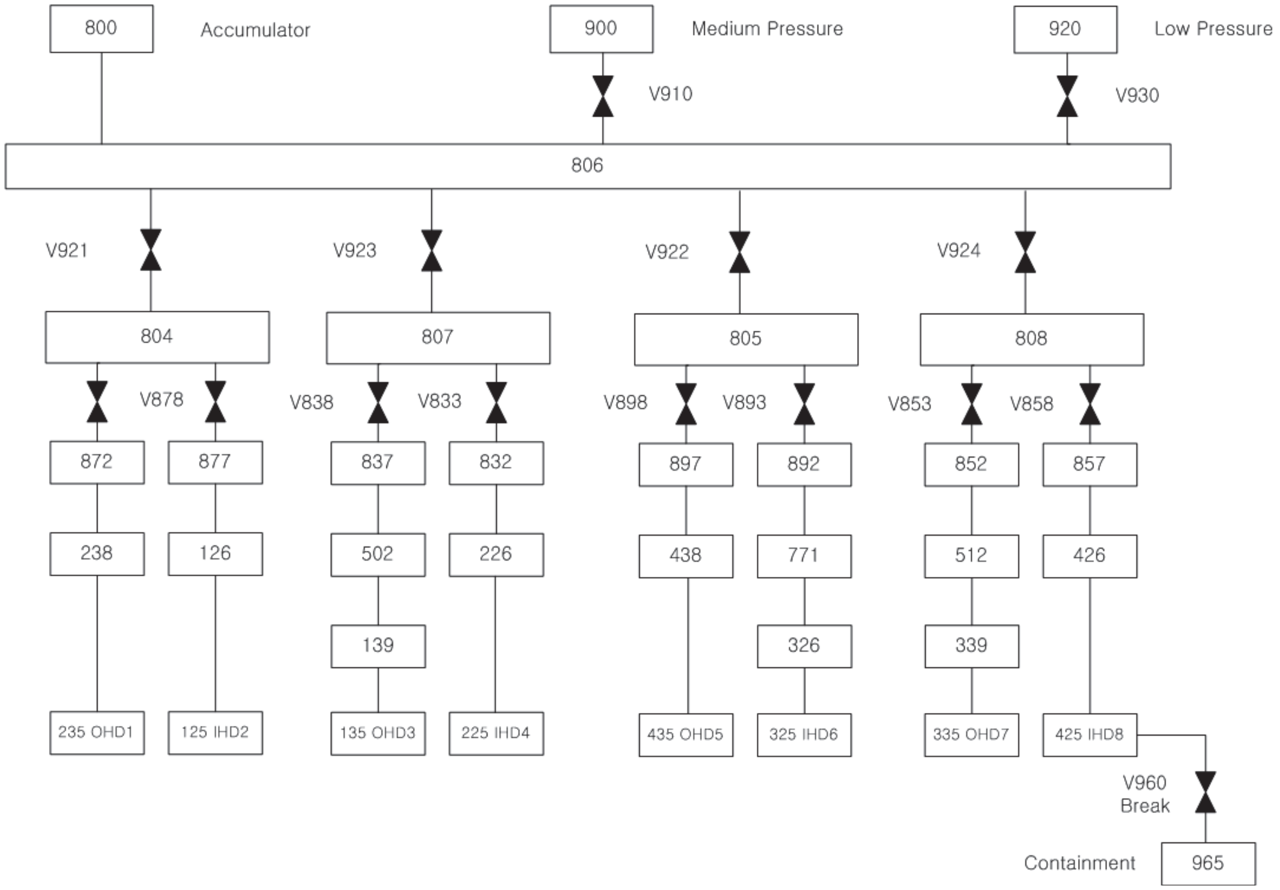
PT-CT Heat Structure

Zr-CO2-Zr

Secondary System



ECCS System


800
Accumulator
900
Medium Pressure
920
Low Pressure
V910
V930
806
V921
V923
V922
V924
804
807
805
808
V878
V838
V833
V898
V893
V853
V858
872
877
837
832
897
892
852
857
238
126
502
226
438
771
512
426
139
326
339
235 OHD1
125 IHD2
135 OHD3
225 IHD4
435 OHD5
325 IHD6
335 OHD7
425 IHD8
V960
Break
Containment
965

III. Uncertainty Parameter Selection

PIRT By Specialist

- Reference [1], [4], [5]
- 2010.10.14 PIRT Meeting

PIRT Sample for CANDU 35% RIH Break Table

H : Modeling+Uncertainty

M : Modeling only

L : Modeling not required

No.	System	Component	Process/Phenomenon	Ranking by Time Phase			Remarks
				1	2	3	
1-3	Fuel Bundle	Structure	Stored energy release	H	M	M	Cp-pellet,Cp_clad,initial power, initial temperature
1-4	Fuel Bundle	Structure	Conduction	H	M	M	K_pellet, K_clad
1-6	Fuel Bundle	Structure	Gap conductance	H	M	L	gap conductance
1-7	Fuel Bundle	Structure	Forced convection to liquid	H	M	M	HTC_liquid_cov
1-8	Fuel Bundle	Structure	Boiling - nucleate	H	M	M	HTC_nb
1-9	Fuel Bundle	Structure	Boiling - film	H	M	L	HTC_film
1-11	Fuel Bundle	Structure	Critical Heat Flux (CHF)	H	M	L	CHF
1-12	Fuel Bundle	Structure	Oxidation	H	L	L	Metal-water reaction model
1-21	Fuel Bundle	Structure	Decay heating	H	M	M	decay heat model
1-22	Fuel Bundle	Structure	Fission heating	H	M	M	fission power model(table)
1-23	Fuel Bundle	Structure	Reactivity - fuel temp.	H	L	L	FTC
1-26	Fuel Bundle	Structure	Reactivity - density/void	H	L	L	MDC(moderator density curve)/MTC(moderator temp. coef.)
1-27	Fuel Bundle	Structure	Radial power distribution	H	M	M	power distribution(heat structure model)
1-36	Fuel Bundle	Volume	Stratification	H	M	L	Include Conservative Model
3-2	Shutdown Systems (SDS)	SDS1(Shut-off Rods)	Reactivity effect of firing SDS1	H	L	L	power table
3-3	Shutdown Systems (SDS)	SDS2(Poison Injection)	Reactivity effect of firing SDS2	H	L	L	power table
4-17	Heat Transport System	Flow Path	Flow- critical flow	H	L	L	critical flow model

Parameter Selection

Based on PIRT

Consider :
1. LOCA break size
2. LOCA break development time
3. Gap size
4. Pump two phase performance

Pending:
1. Reflood Heat transfer variables
2. Initial Power and Temperature related variables
   -Too much uncertainty in nodal, radial, axial distribution

Input Parameters

No	Variables	PDF	Min	Max
1	Initial Oxide Thickness	uniform	2.00E-06	4.50E-05
2	Cp Pellet Mult.	normal	0.9	1.1
3	Cp Clad Mult.	normal	0.9	1.1
4	K Pellet Mult.	normal	0.9	1.1
5	K Clad Mult.	normal	0.9	1.1
6	Pump Two Phase Mult.	uniform	0	1
7	Break Area Mult.	uniform	0.7	1.15
8	Break Develop Time.	uniform	0	1
9	Disc. Coef. for HF Crit. Flow	normal	0.9	1.1
10	Thermal non-Eq. const. for HF Crit. Flow	normal	0.1	0.18

IV. Analysis

Basic Assumption of Analysis

LOCA Acceptance Criteria

3.1 Steady-State Analysis

3.2 Transient Analysis

3.3 Input Preparation for Uncertainty Analysis

3.4 Results of Runs

Basic Assumption of Analysis

- The reactor trips immediately after initiation of break
- LOCA signal : P_IHD-MAX(MIN(P_OHD)) < 5.25MPa
- Feed and bleed system, the PZR isolation after LOCA
- PHT pumps trip after a delay of 5 sec after LOCA
- governor valve closes and FW stops after 5 sec from LOCA
- MSSVs are credited for steam generator crash cool down and are opened after a delay of 30 sec from LOCA
- AFW starts 3 minutes after MFW trip
- RRS is assumed frozen
- HPECC Injection Signal is generated when P_IHD-MAX(MIN(P_OHD)) < 3.62 Mpa
- MPECC starts after 1.5 min. of the HPECC start
- LPECC starts immediately after the MPECC stops

LOCA Acceptance Criteria

- CANDU LOCA Integrity Acceptance
  - Maximum Centerline Temperature< 2840℃
  - Maximum Clad Temperature <1760℃
  - Fuel Channel Failure by Internal Overheating
    - Pressure Tube Temperature < 600℃
  - Clad Temperature < 800℃ (Dry-out Condition)
  - Embrittlement mechanisms of Zr alloy
    - Maximum Clad Temperature < 1200 ℃

- Limitations
  - Direct Application of Clad Temperature
  - Fixed Power Distribution
  - Did not consider Reactivity Model

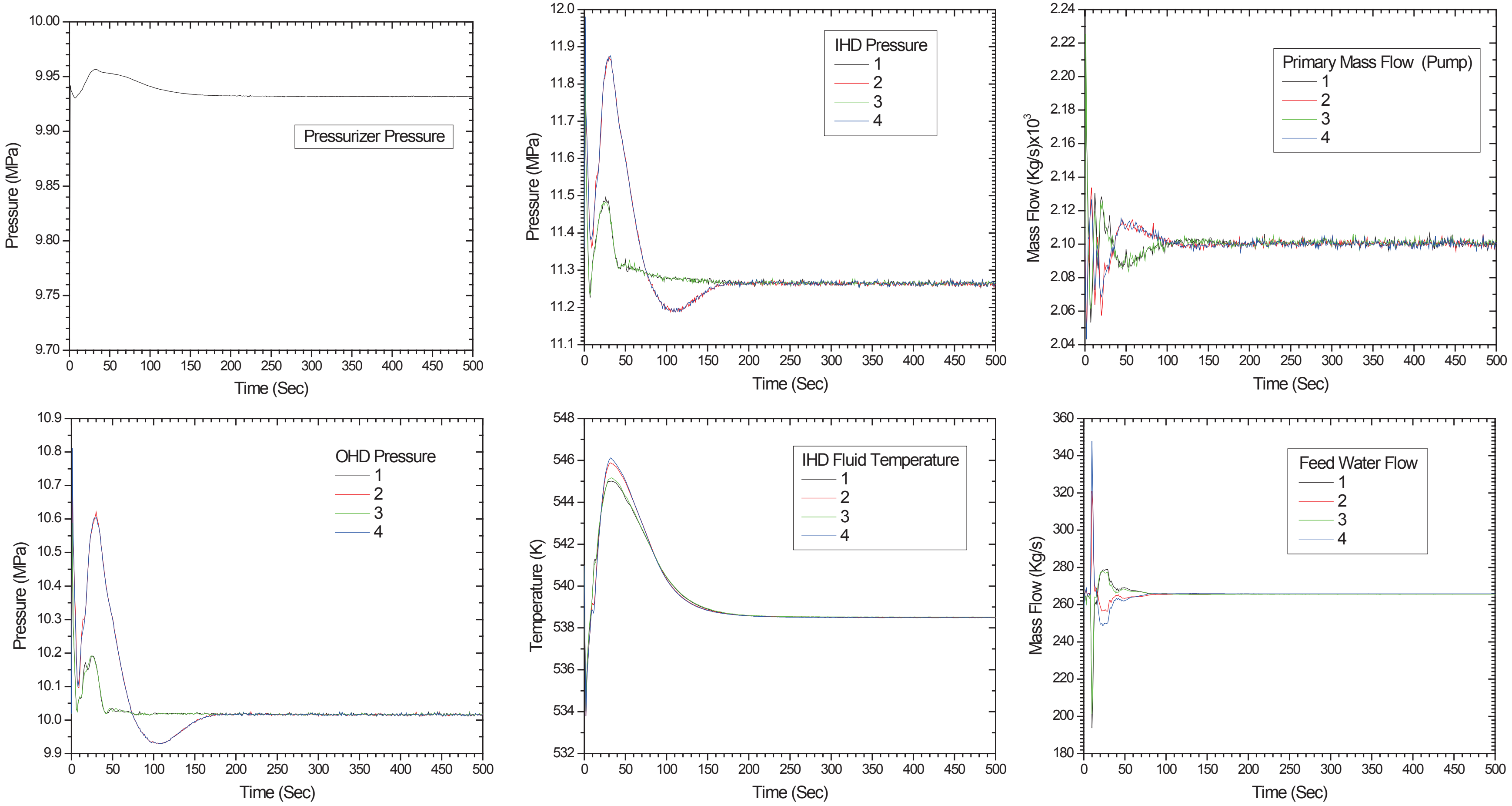
3.1 Steady-State Analysis

- Initial Conditions

	Parameters	Unit	Value
1	Inlet Header Pressure	MPa	11.266
2	Outlet header Pressure	MPa	10.017
3	Pump Discharge	Kg/s	2100.2
4	SG Inlet Plenum fluid temperature	K	583.14
5	SG Outlet Plenum Fluid temperature	K	537.85
6	Core Inlet mass Flow	Kg/s	2100.1
7	SG Steam Flow	Kg/s	265.12
8	SG Level	m	11.833
9	SG Dome Pressure	MPa	4.7706
10	SG Power	MW	530.96

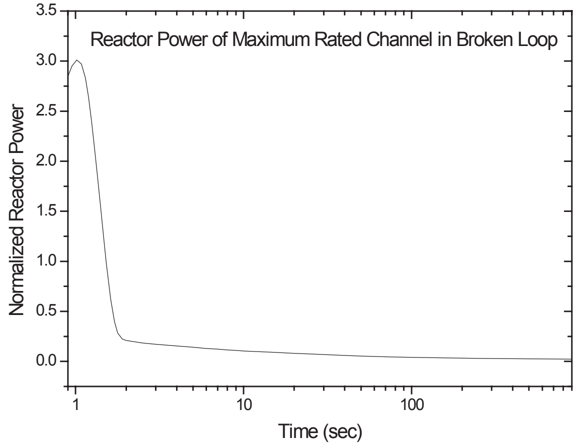
Transient : 35% RIH Break Occurs at time 0 sec

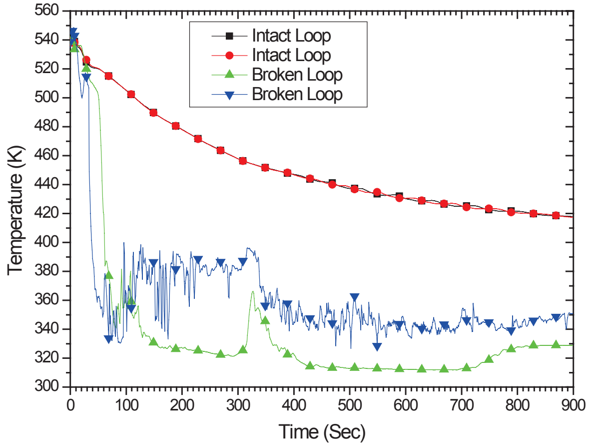
- Fully Matches with Utility Results(CATHENA)

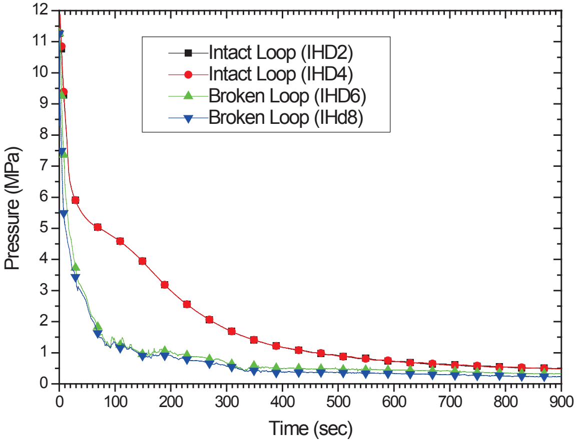


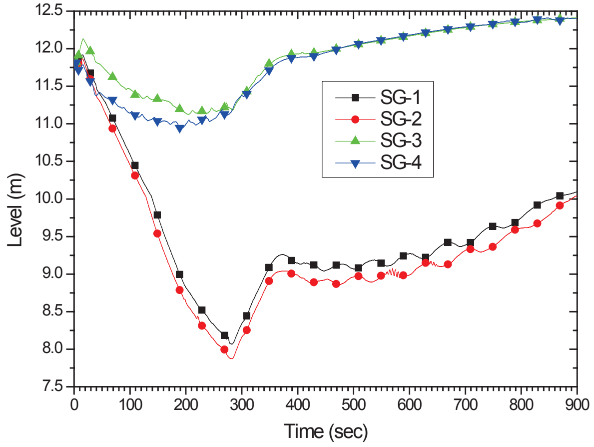
3.2 Transient Analysis

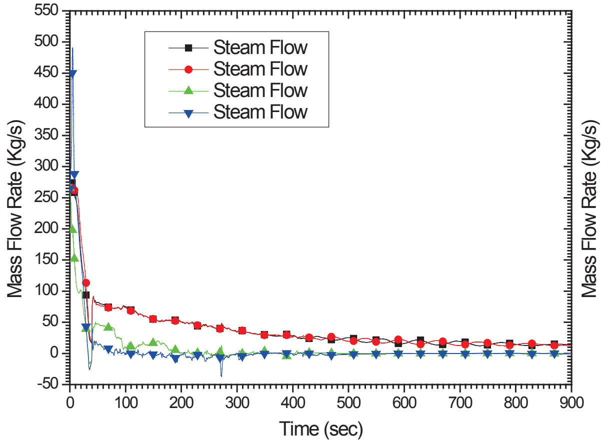
- Shows Nearly the same Results as in FSAR

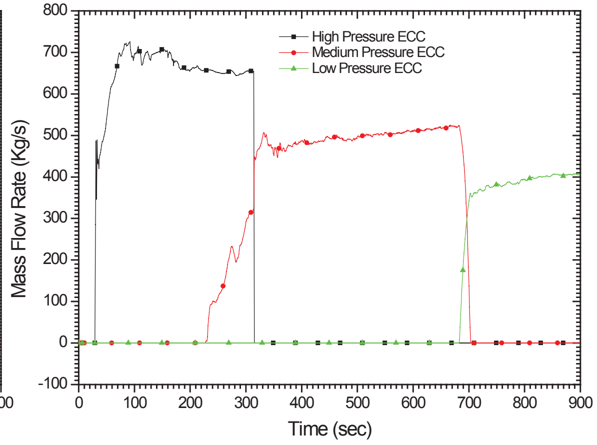












SOE of LOCA

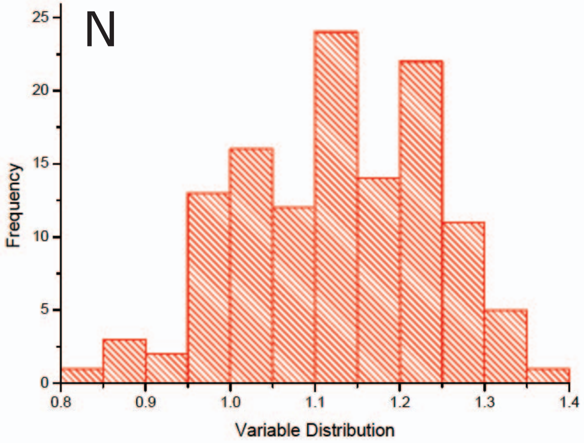
Time (sec)	Event
0	35% RIH Break Occurs Reactor Trips
9.07	LOCA signal generated Feed & Bleed system isolated Pressurizer system isolated ECCS valves open
14.08	Turbine rolls back. Main Feed water pump stops PHT Pumps trip
18	Clad surface temperature reaches maximum value of 1022C
30	High Pressure Injection Signal generated High Pressure Injection starts
120	Medium pressure injection signal generated
194	Auxiliary Feed Water Pump starts
220	Medium Pressure Coolant injected into core
314	High pressure injection stops
670	Medium Pressure Injection stops Low pressure injection starts

3.3 Input Preparation for Uncertainty Analysis

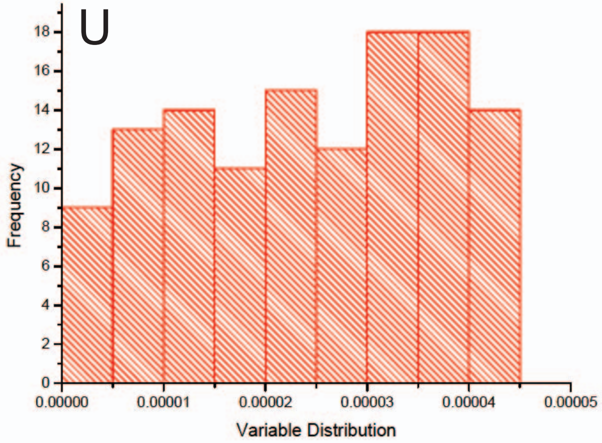
- LHS Sampling Program : KINS In-house Utility
-12Input Parameter Distribution: 124 Sample Cases
    -95% tolerance, 95% confidence
    -Wilks third order

No	Th. Ox.	Cp pellet	Cp clad	K pellet	K clad	PMP Mlt.	A break	T break	Cd	Cne
1	2.92E-05	0.971221	0.96745	0.994894	1.031247	0.173547	1.045828	0.358937	0.993123	0.135451
2	4.40E-05	0.962406	0.943264	0.963041	1.019421	0.411588	0.850418	0.570963	0.993239	0.155691
3	3.15E-05	0.999045	1.044174	1.015516	0.964521	0.771209	1.119055	0.901058	0.936834	0.131578
4	2.06E-05	0.996834	0.97774	1.021073	1.007909	0.181583	1.109579	0.231296	0.978049	0.16971
5	3.97E-05	1.005733	0.99653	0.955392	0.990567	0.836577	0.969681	0.959822	0.995166	0.131705
6	3.96E-05	0.97741	0.987822	0.971512	1.007233	0.736669	1.111351	0.457067	1.012038	0.144615
7	3.96E-05	1.013469	1.022497	0.991028	1.049413	0.508812	0.95764	0.902551	0.976986	0.137486
8	2.18E-05	1.040008	0.984789	1.077331	0.962522	0.781878	1.079683	0.899328	1.005041	0.134268
9	1.47E-05	1.014303	0.983103	1.014845	1.042251	0.746736	0.841023	0.806899	1.047783	0.152318
10	1.83E-05	1.041304	1.016778	0.96603	1.004636	0.945246	0.713894	0.414068	1.016961	0.14507
...	...	...	...	...	...	...	...	...	...	...
124	2.34E-05	0.999342	0.977623	1.001765	1.015436	0.285681	1.125241	0.480152	1.059111	0.137259

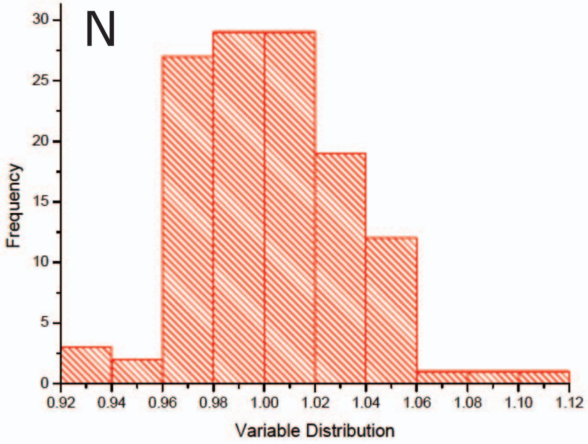
Code Input Distribution



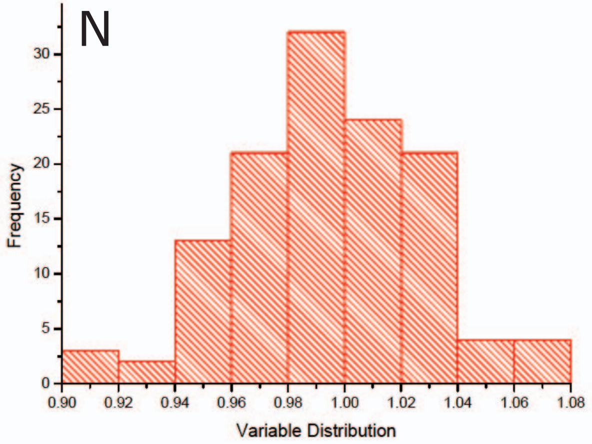
Gap. Cond



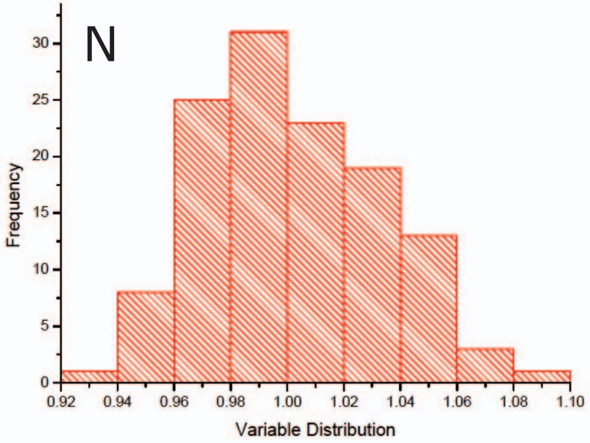
Oxide Thickness



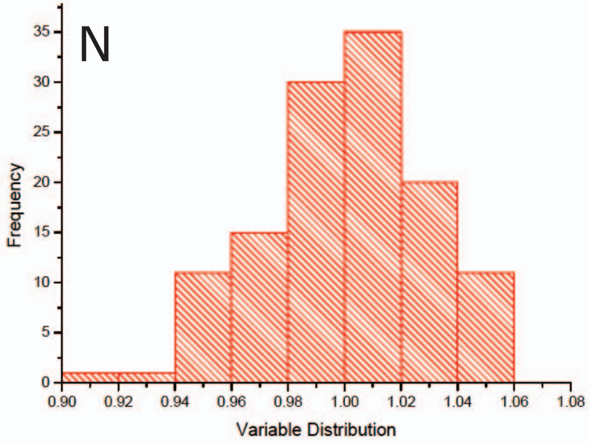
Pellet CP



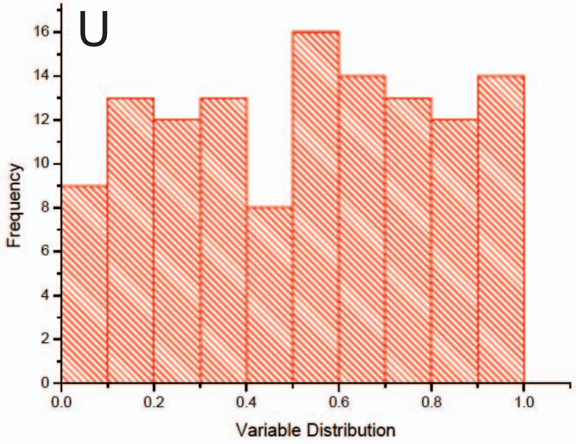
Clad CP



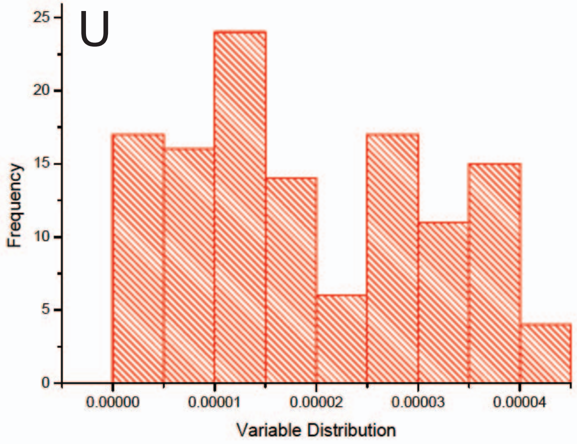
Pellet K



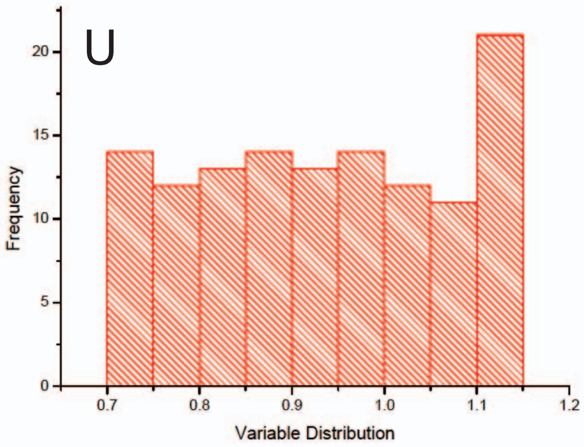
Clad K



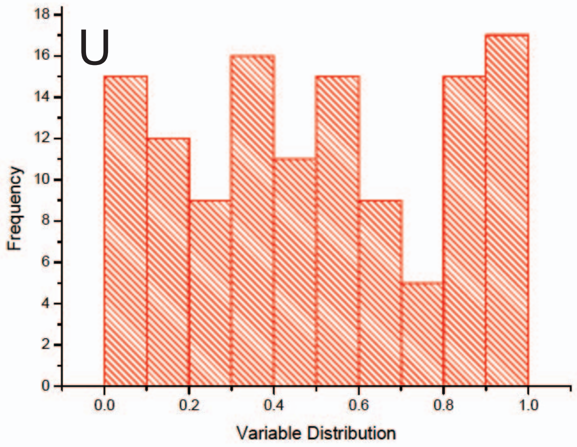
Pump 2 phase mult



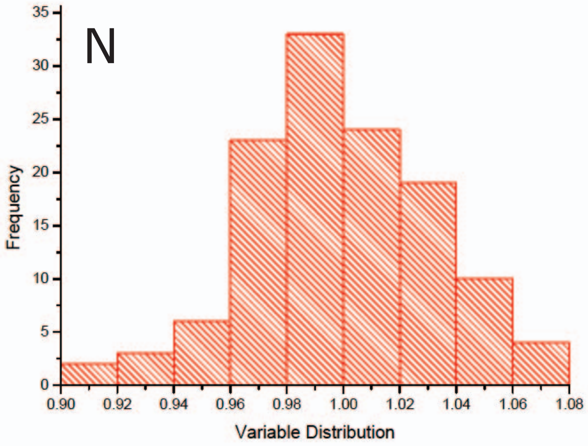
Cold Gap size



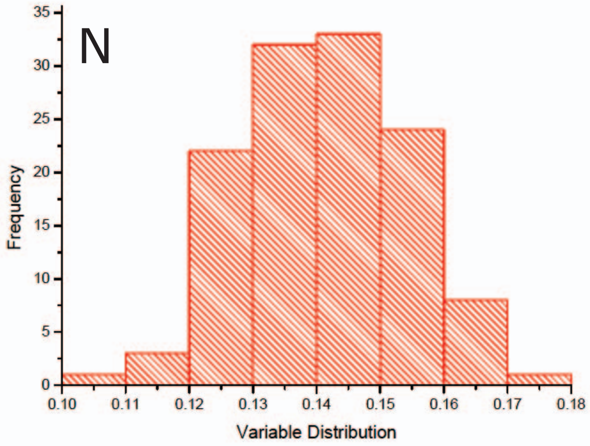
Break A mult



Break time



Disc coef



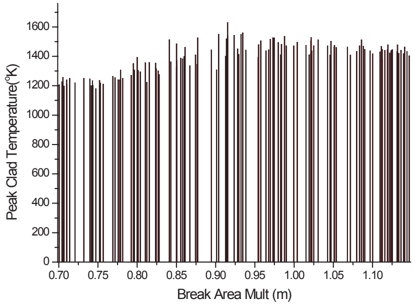
Non eq const

3.4 Results of Runs

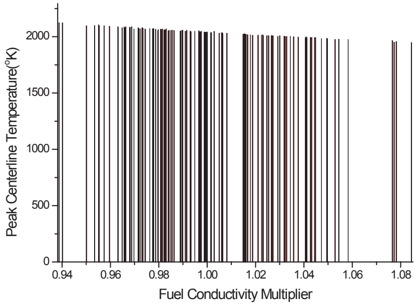
Parameter Dependency

- **Most ParameternShows No Simple Correlations with Results**
  - **Break Area and Fuel Conductivity Shows Simple linear correlations**

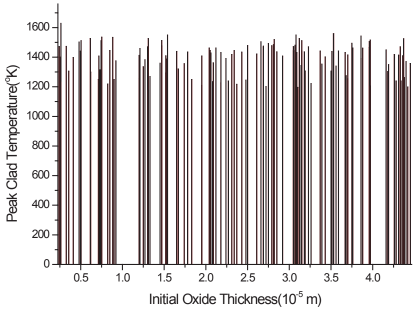
Effect of Break Area
on PCT



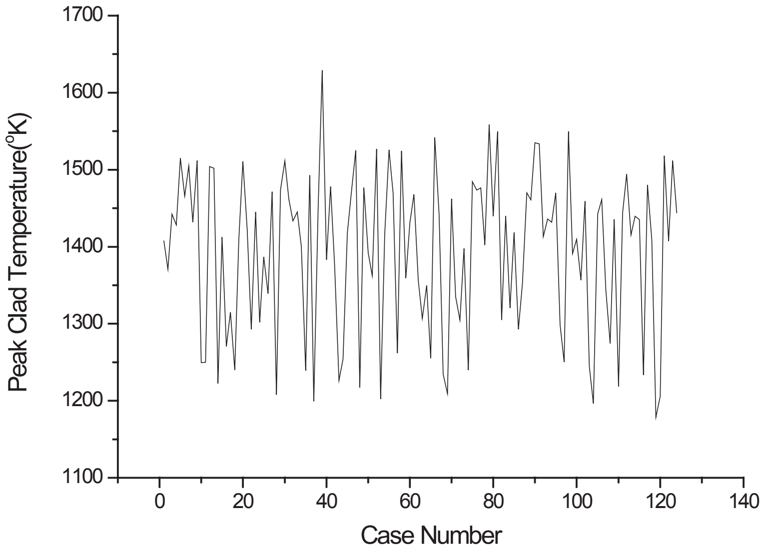
Effect of Fuel Conductivity
On Max. Fuel Temperature



Typical Effect of Parameters
on PCT

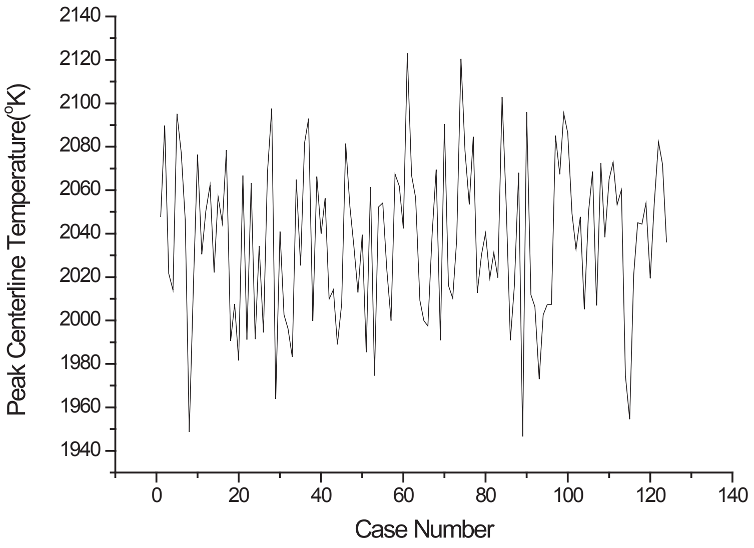


Peak Clad Temperature Rank



Ranking	Case No	PCT(°K)
1	123	1544.35
2	13	1535.28
3	52	1527.02
4	30	1525.95
5	39	1522.61
6	79	1520.58
7	98	1511.17
8	95	1502.14
9	91	1496.83
10	70	1496.66

Peak Center Line Temperature



Ranking	Case No	PCT(°K)
1	61	2155.97
2	74	2153.68
3	84	2136.34
4	28	2130.06
5	90	2128.92
6	5	2127.51
7	99	2126.49
8	37	2124.81
9	2	2123.86
10	70	2123.62

Fuel failure

- **More Fuel Bundles are Damaged than Utility Result (14 Bundles in FSAR)**

Case No.	PCT(°K)	Ruptured Bundle No.	Failed Node(1: Failed, 0: Safe)						
			4-1	4-2	4-3	4-4	4-5	4-6	4-7
13	1535.28	53	000000000000	000000000000	000000100000	000001110000	000000000000	000000000000	000000000000
39	1522.61	48	000001100000	000001100000	000000000000	000000000000	000000000000	000000000000	000000000000
123	1544.35	42	000000000000	000000000000	000000000000	000001110000	000000000000	000000000000	000000000000
52	1527.02	36	000000000000	000000000000	000001100000	000000100000	000000000000	000000000000	000000000000
30	1525.95	28	000000000000	000000000000	000000000000	000001100000	000000000000	000000000000	000000000000
70	1496.66	28	000000000000	000000000000	000000000000	000001100000	000000000000	000000000000	000000000000
76	1496.07	28	000000000000	000000000000	000000000000	000001100000	000000000000	000000000000	000000000000
88	1478.53	25	000000000000	000000000000	000000100000	000000100000	000000000000	000000000000	000000000000
5	1493.18	24	000000000000	000001100000	000000000000	000000000000	000000000000	000000000000	000000000000
58	1488.28	24	000000000000	000001100000	000000000000	000000000000	000000000000	000000000000	000000000000
66	1494.1	24	000000000000	000001100000	000000000000	000000000000	000000000000	000000000000	000000000000
79	1520.58	24	000000000000	000001100000	000000000000	000000000000	000000000000	000000000000	000000000000

V. Conclusions & Remarks

Conclusions

Ongoing Works

References

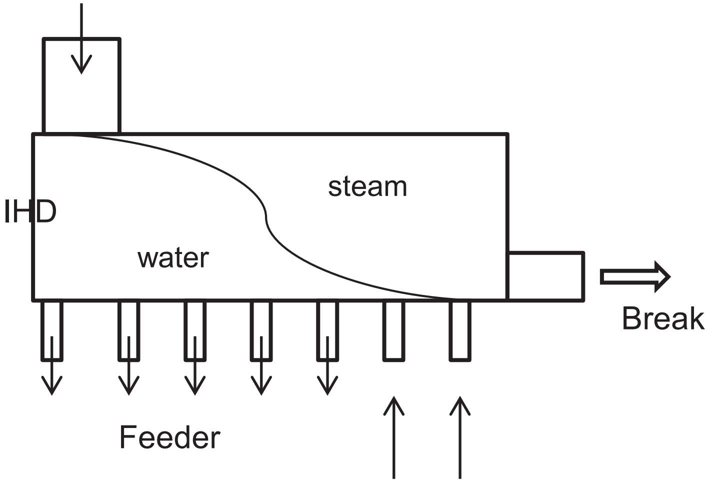
Conclusions

- MARS simulates CANDU Transient well
- Shows Acceptable Safety Analysis Results
  - More conservative Fuel Failure results
- Evaluation of Uncertainty Parameter Should be done after
  - Power & Reactivity Parameter uncertainty Models are Considered
- Note
  - Also MARS-SCAN Methodology was Developed in Parallel

Ongoing Works

- Considering Positive CVR
  - Tests and Experiments of Uncertainty/Importance of CVR
  - Need to Quantify CVR Effect

- Evaluation of Safety Features of System
  - Trips, Delays
  - System Models
    - ✓ T/H
    - ✓ Kinetics
      - ✓ Part Core Kinetics
      - ✓ Multi Core Simulations



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Thank You

Advanced approach to consider aleatory and epistemic uncertainties for integral accident simulations

Jörg Peschke, Martina Kloos
GRS

November 16-18, 2011
OECD/NEA Workshop, Barcelona, Spain

Content

1. Motivation
2. MCDET – A Probabilistic Dynamics Method
3. Crew Module – Simulating Human Actions
4. Example Application – Station Black Out
5. Conclusions

Motivation for Probabilistic Dynamics Methods (1)

- Deterministic codes are increasingly used for risk informed decision making, safety analyses and licensing procedures.
- Current trend: Using Best Estimate codes with additional epistemic uncertainty analysis of model input parameter.
- Substituting conservative with more realistic input data accounting for their epistemic uncertainties allows a
  - ➢ quantification of the uncertainties of the code prediction and
  - ➢ more precise assessment of safety margins.
- Up to this point there is no need for Probabilistic Dynamics methods.
- Is it sufficient to consider only epistemic uncertainties ?

Motivation for Probabilistic Dynamics Methods /2/

- Any process simulated by a deterministic code and used for risk informed decision making is subjected to aleatory uncertainties (stochastic influences).
- The evolution of nonlinear dynamic processes might be sensitive to small deviations in process conditions.
- Random events do not only influence the process evolution, but the process might also affect the stochastic behavior of random events (dynamic-stochastic interaction).
- To model a dynamic process as realistic as possible and to get a more detailed insight of the process behavior
  - the influence of aleatory uncertainties and
  - the dynamic–stochastic interactions

  must explicitly be taken into account.
- Probabilistic Dynamics methods try to satisfy these aspects by combining deterministic and probabilistic issues in a consistent manner.

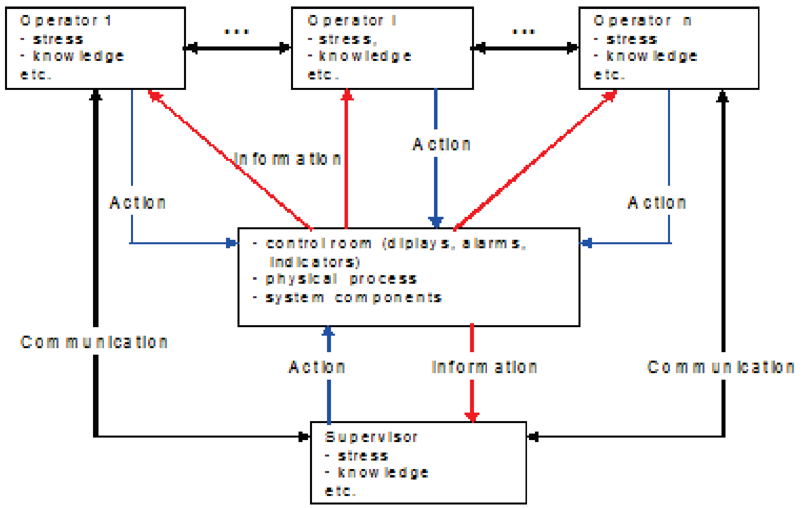
MCDET - Monte Carlo Dynamic Event Tree

- The method MCDET was developed at GRS to perform a Probabilistic Dynamics Analysis.
- Characteristics of the MCDET-method:
  - The MCDET-method allows to model any aleatory and epistemic uncertainty with any probability distribution (discrete, continuous).
  - The MCDET-method was implemented as a flexible module which can be combined with any deterministic code.
  - The combination of the MCDET-module with a deterministic code automatically provides a sample of DDETs with varying times and order of random events.
  - Data resulting from a MCDET-analysis can be analyzed by statistical methods providing probabilistic information about the concerning process.
  - The MCDET-method avoids drawbacks of MC simulation and the DDET approach if they are separately applied.

Crew-Module (1)

- The MCDET-module was supplemented by a Crew-module to account for human actions within an MCDET-analysis.
- Simulation of operator actions as a dynamic process
    - evolving in parallel and interacting with the physical process.

Situation simulated by the Crew – Module:



Operator 1
- stress
- knowledge
etc.

...

Operator i
- stress,
- knowledge
etc.

...

Operator n
- stress
- knowledge
etc.

Information

Action

Action

Action

- control room (diplays, alarms, indicators)
- physical process
- system components

Communication

Action

Information

Communication

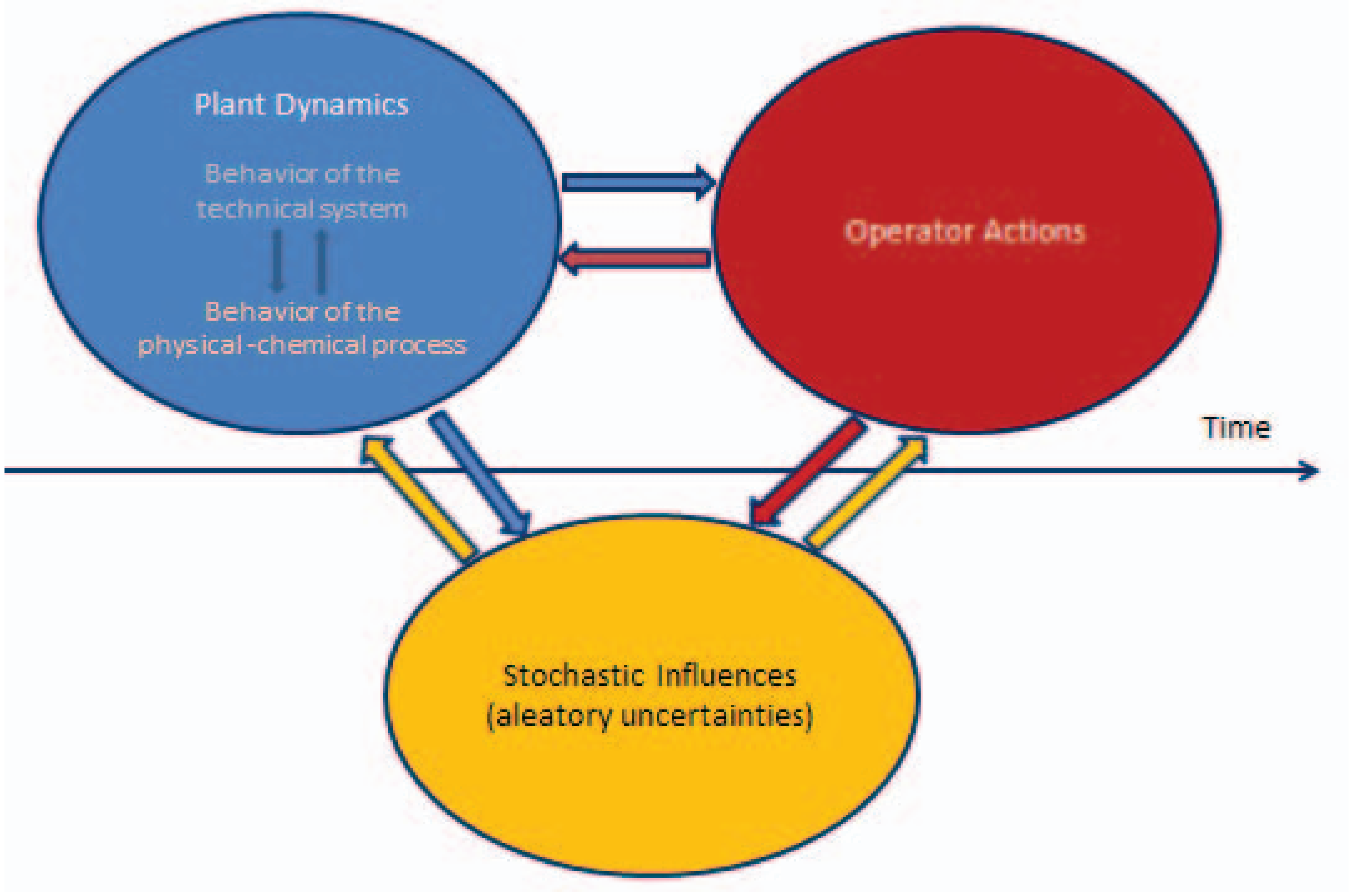
Supervisor
- stress
- knowledge
etc.

- A crew can consist of a number of operators each responsible for several tasks.
- Tasks can be executed in parallel or be started only, if specific conditions are accomplished.
- Crew members can communicate and get information of the process state and system components from the control room displays.
- Actions of operators might cause a change of components states immediately influencing the further process evolution.

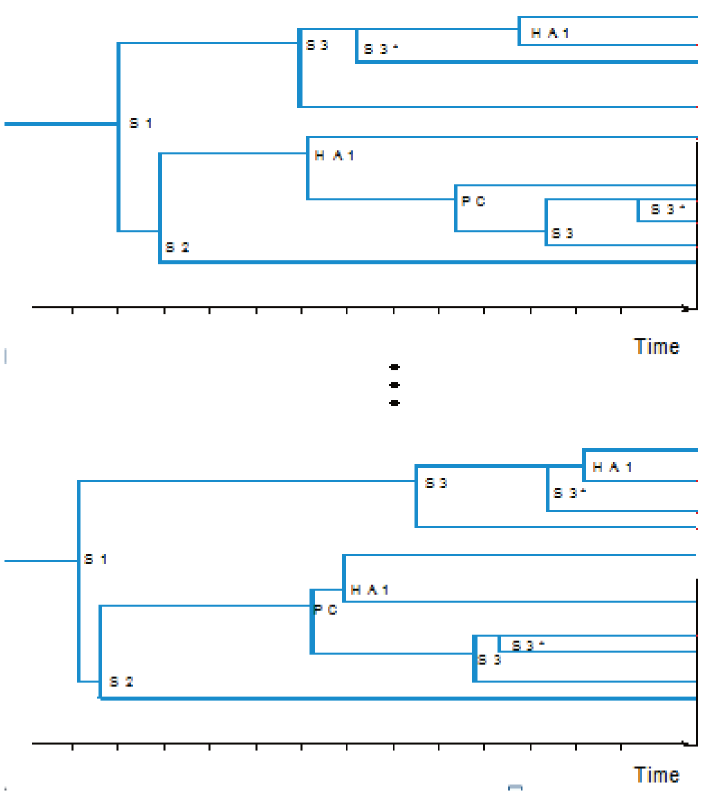
Crew-Module (2)

- Simulating the dynamic process of human actions, stochastic as well as dynamic elements are considered.
  - Stochastic elements:
    - execution time of actions,
    - human error probabilities (errors of omission, errors of commission).
  - Dynamic elements are e.g.:
    - monitoring tasks,
    - duration up to the time a special condition takes place,
    - exchange of information through communication,
    - increase or decrease of the stress level of operators depending on system state.

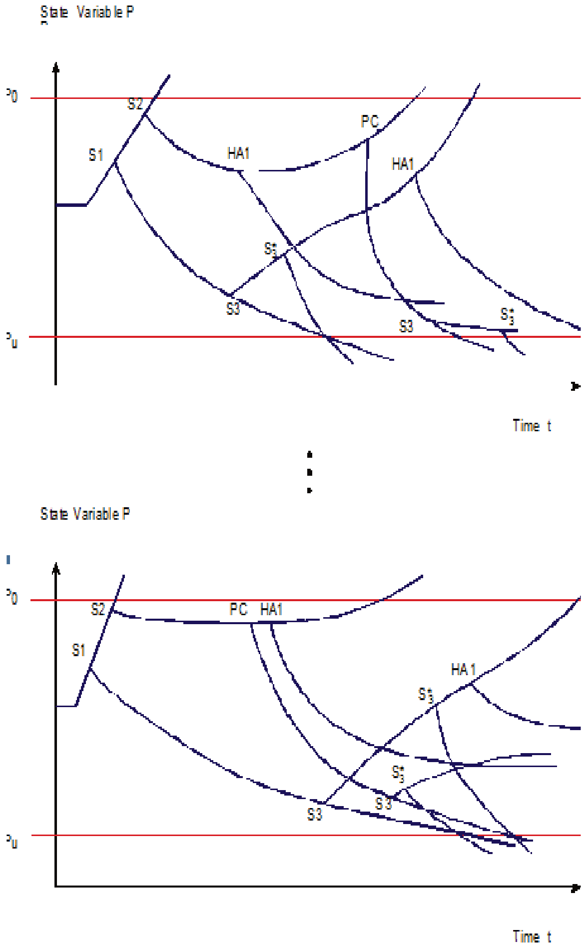
MCDET coupled with a deterministic code allows an integral simulation of complex dynamic systems where interactions of the technical system, the physical process, human actions and influences of stochastic quantities (aleatory uncertainties) are taken into account in the course of time.



The Output of an MCDET analysis is a sample of DDETs in the Time/Event space as well as in the Time/State space



Time/Event space



Time /State space

Connected to each sequence in the Time/Event space is :

- the corresponding sequence in the Time/State space for any process quantity
- the probability of the corresponding sequence

To each DDET the conditional probability distributions of any process quantity can be calculated.

The final result is the mean (mixed) probability distribution over all generated DDETs with corresponding confidence limits indicating the sampling error.

Application – Station Black-out scenario (1)

Assumptions:

- Total loss of power, i.e. power from offsite systems, emergency diesels, main coolant pumps and all other operational systems are not available.
- Batteries guarantee a power supply for some period of time.
- Scram and turbine trip are automatically performed.

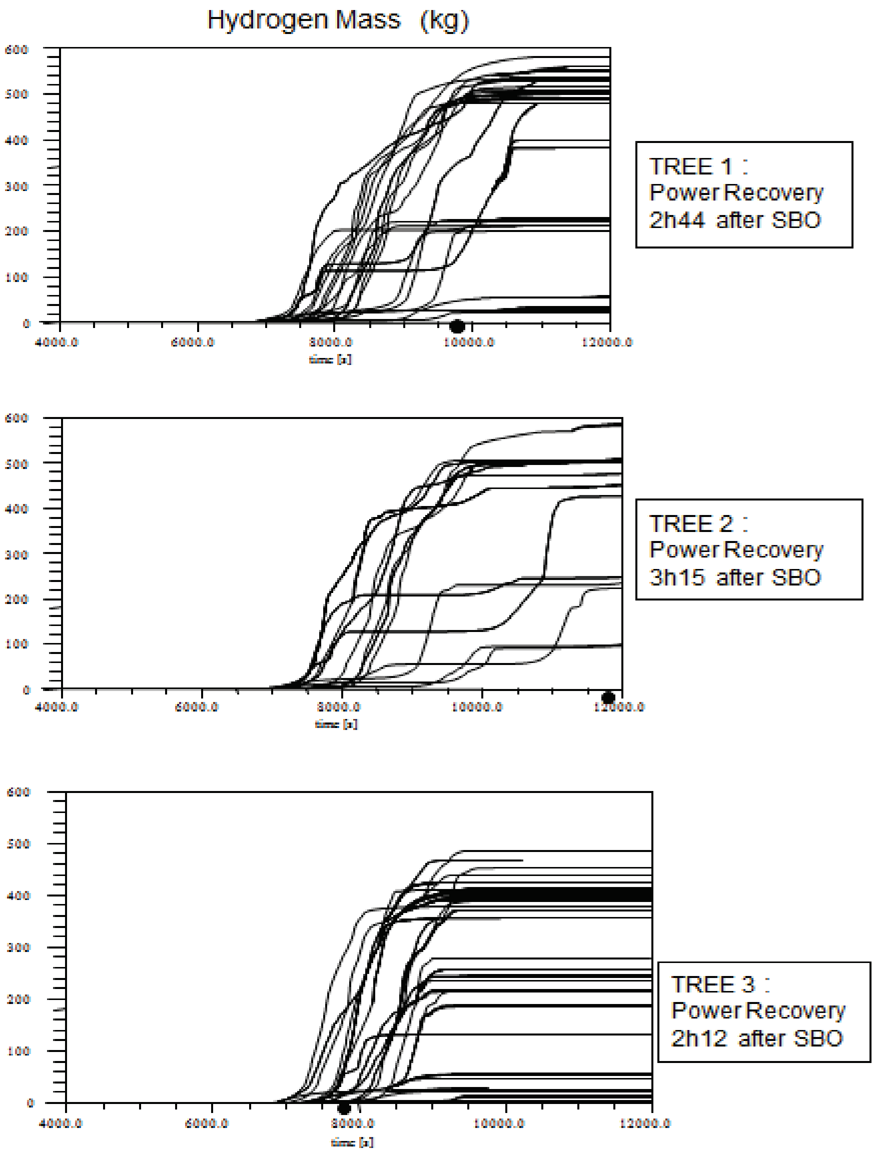
Aleatory uncertainties relate to

- the opening and closing function of the PORV and 2 SV during automatic pressure limitation (failure probabilities depend on the no. of demand cycles),
- the demand cycle (time) at which the pressurizer valves may fail,
- the time needed for primary side pressure release,
- the opening function of the accumulator valves on demand,
- the time of power recovery (only then the ECCS can be reconnected),
- the availability and times when high and low pressure pumps of ECCS are reconnected to the grid,
- the opening function of the source isolation valves.

Application – Station Black-out scenario (2)

- Epistemic uncertainties refer to the probability models of the aleatory uncertainties.
- Epistemic uncertainties were considered by their reference values.
- The MELCOR/ MCDET combination was used to generate a sample of 50 DDETs.
- Event sequences were calculated up to 12000 s.
- The available information for each sequence comprise:
  - the order of events and the points in time when stochastic events occur,
  - the occurrence probability and
  - the time histories of 80 dynamic quantities (e.g. H_2 mass, UO_2 melt mass, temperature and pressure in the RPV, cladding temperature etc.).
- The data of the MELCOR/MCDET-analysis were analyzed with statistical methods.
- Probability distributions were derived together with corresponding 90%-confidence intervals.

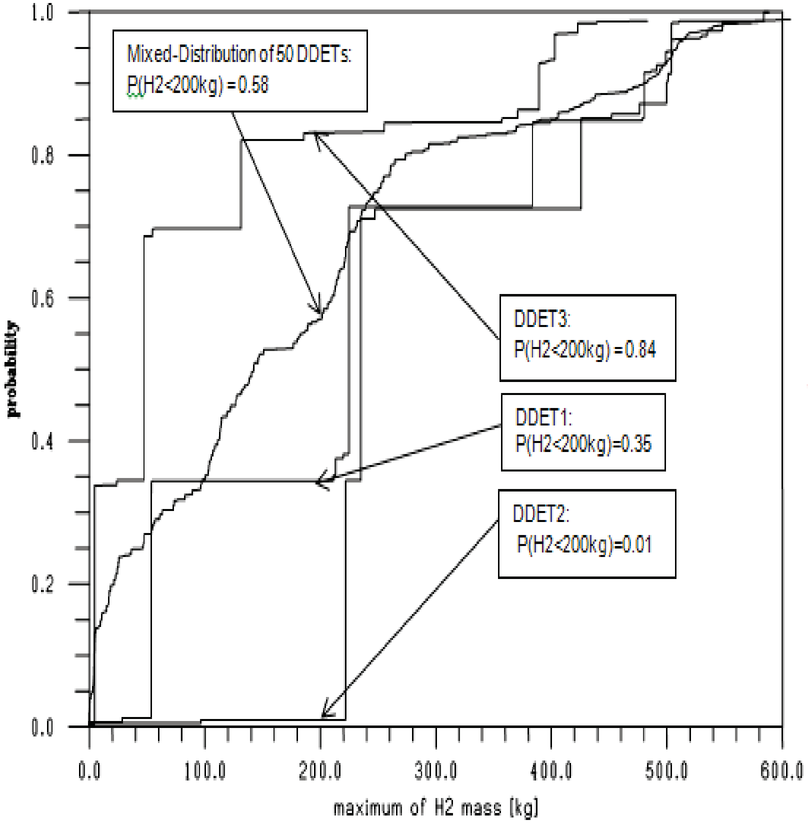
Application – Station Black-out scenario (3)

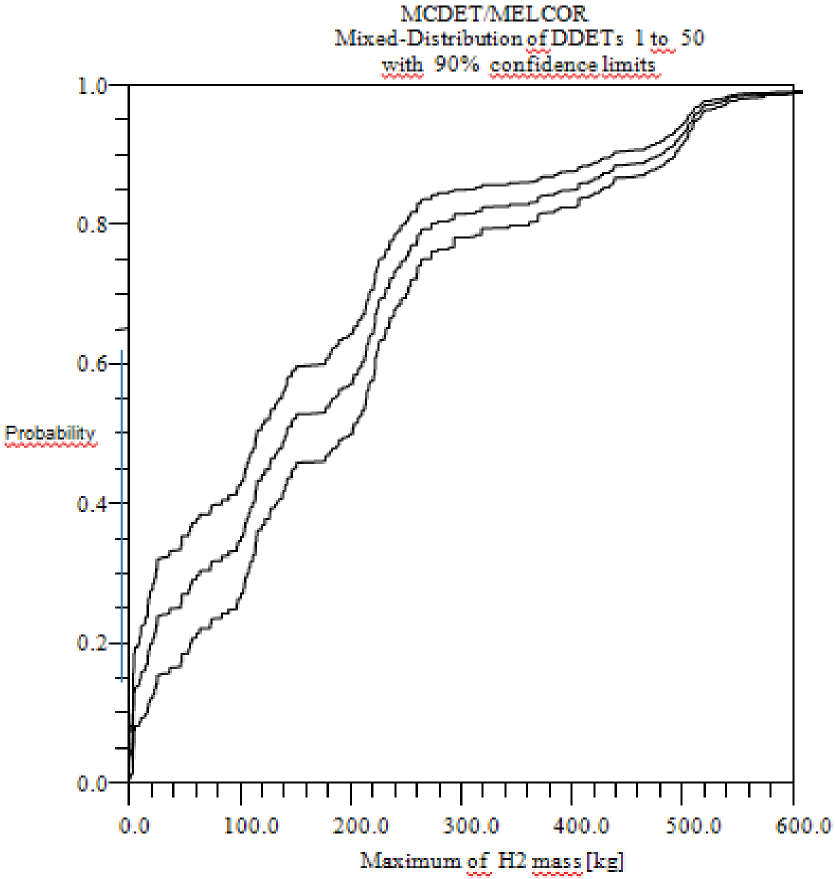


- The variability within each DDET arise from the discrete variables handled by the DDET approach.
- The variability between the DDETs arise from the continuous variables considered with MC simulation.
- Attached to each sequence are the corresponding events and the occurrence probability.
- The probabilities are used to derive probabilistic assessments of the sequences and their consequences.

Application – Station Black-out scenario (4)

- The probabilistic information of a DDET is the probability distribution of the concerning output quantity, e.g. the maximum of produced H2.





Conclusions (1)

- MCDET is a probabilistic dynamics method which is capable of accounting for any aleatory – and epistemic - uncertainty during an accident scenario.
- The MCDET- and Crew modules can in principal be coupled with any deterministic code simulating a dynamic process.
- The combination of the MCDET- and Crew modules with a deterministic dynamics code
  - allows to perform an integral simulation of a complex system where interactions between the physical process, system behaviour, the process of human actions and stochastic influence factors are taken into account,
  - avoids unnecessary modeling simplifications and hence reduces model uncertainties,
  - provides more detailed insights into process characteristics which might be important for risk-informed decision making.

Conclusions (2)

- The extent to which models are able to optimize decisions regarding system safety depends on
  - the level of detail of the model,
  - its completeness by integrating aleatory and epistemic uncertainties and
  - that assumptions are made as realistic as possible.
- To satisfy these requirements the MCDET-method has been developed to combine deterministic and probabilistic issues in a consistent manner.

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PSI

Wir schaffen Wissen – heute für morgen

Paul Scherrer Institut
Martin A. Zimmermann

Safety Margin Assessment (SM2A): Stimulation for Further Development of BEPU Approaches?

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Content

- Short description of SMAP framework
- Implementation by SM2A task group
- Highlights from SM2A results
- Achievements / Lessons learned
- Recommendations / Further R&D work
- Conclusion

Acknowledgments

The work was performed in the framework of the CSNI Expert Group Safety Margin Assessment and Application (SM2A) by (listed alphabetically according to country and organisation)

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Risk results from a Swiss BWR power up-rate project

Power up-rate of 14.7 % has no significant effect

- Source term changes due to
  - Increase of inventory (proportional to power increase)
  - More rapid sequences
- Risk increases by 25% to 30 %
  - Including risk reduction due to
    - Containment venting system
    - Enlarged sump suction strainers
  - Big uncertainty related to this estimate!
- Erosion of margin possible

What is the problem?

How can a possible erosion of margins to safety be measured that could arise as a consequence of

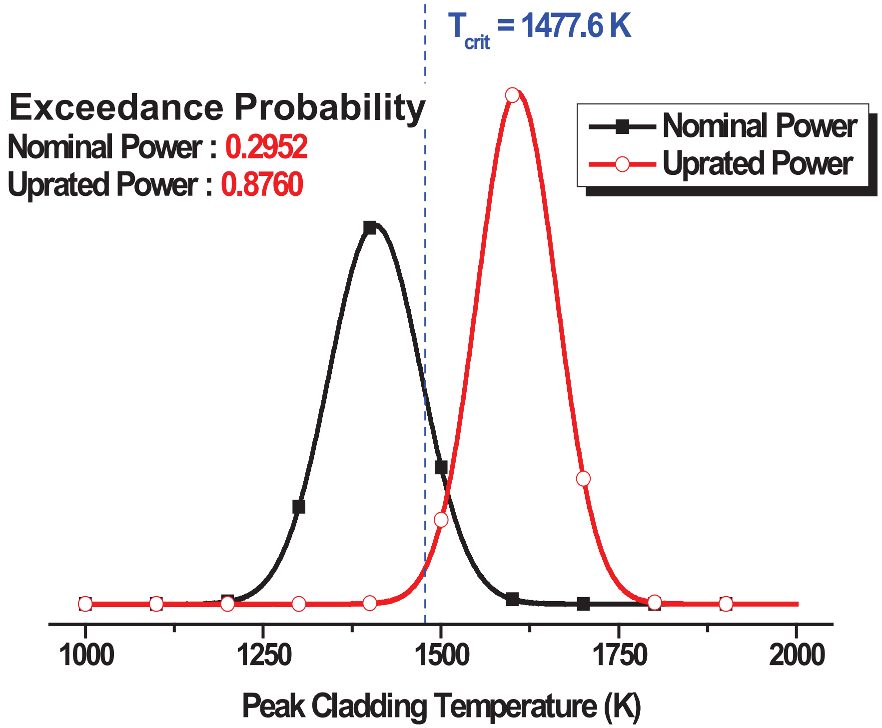
- significant (and ev. multiple) plant modifications
  - both beneficial as well as detrimental

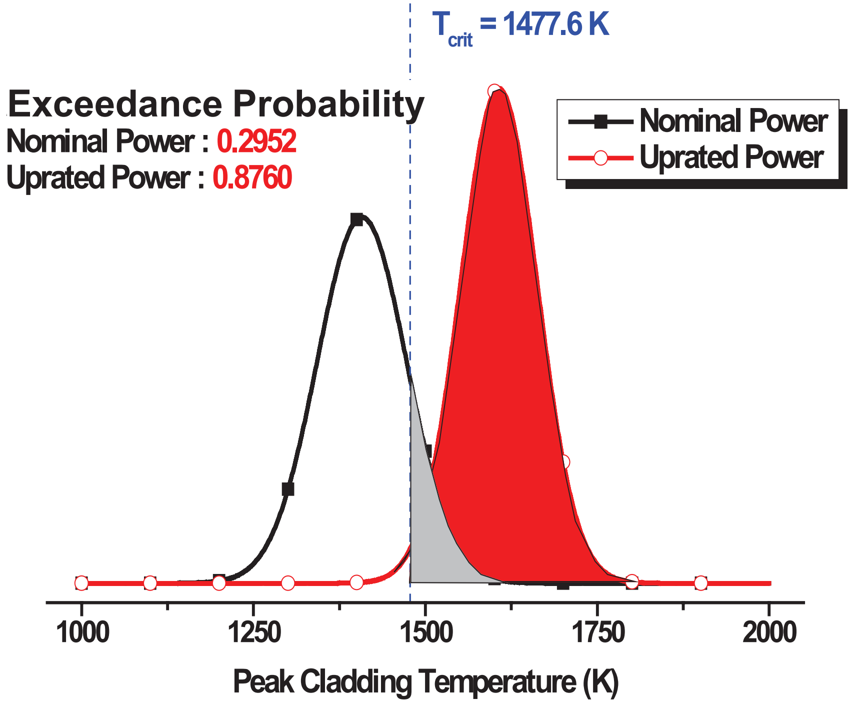
Keeping in mind

- licensing procedure ensures that all the acceptance criteria are observed

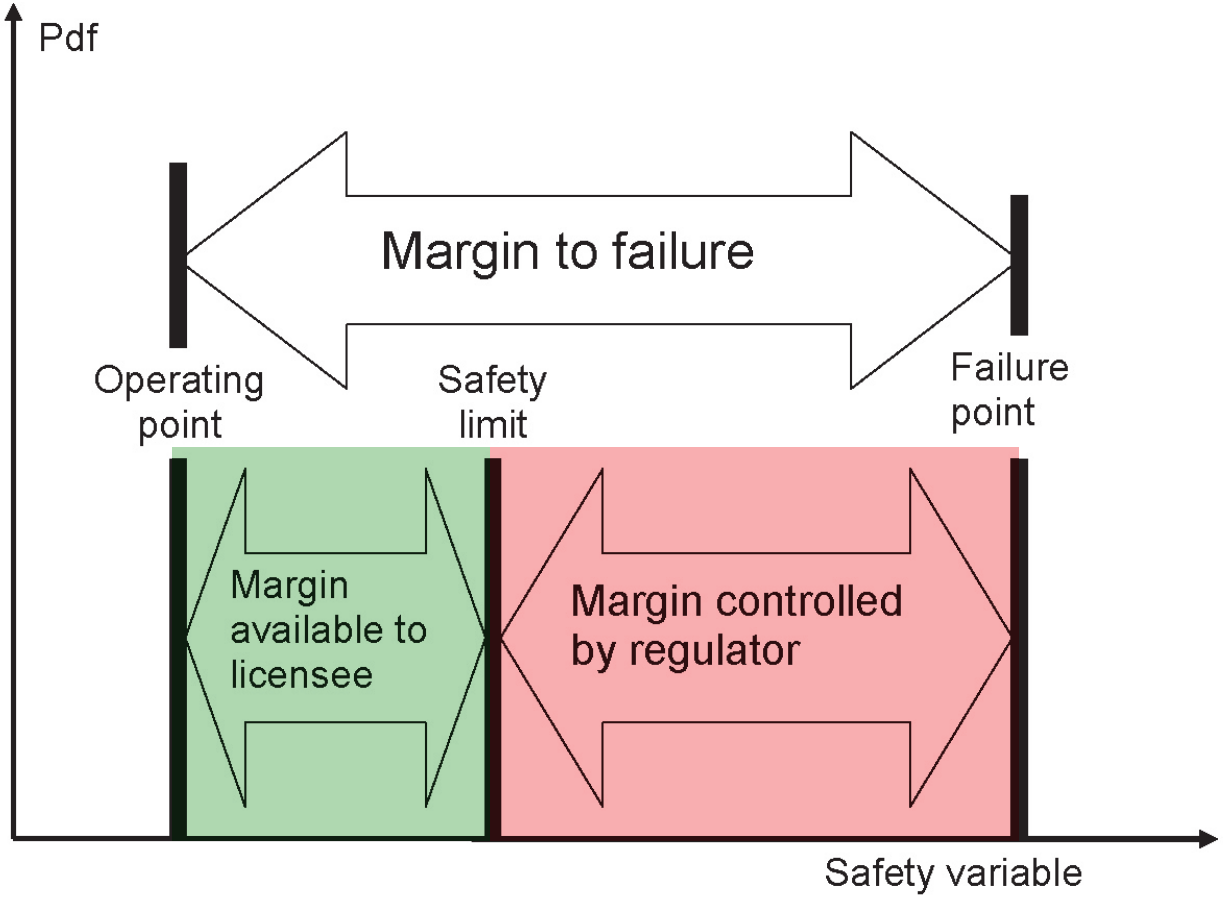
Shift of PDF for MLOCA (6-inch)

- Capacity – Load model of reliability theory
- For SM2A: Capacity is defined by acceptance criterion
- Probability of exceedance





Safety Margin Terminology



Distinguish between margin available to licensee and margin controlled by regulator

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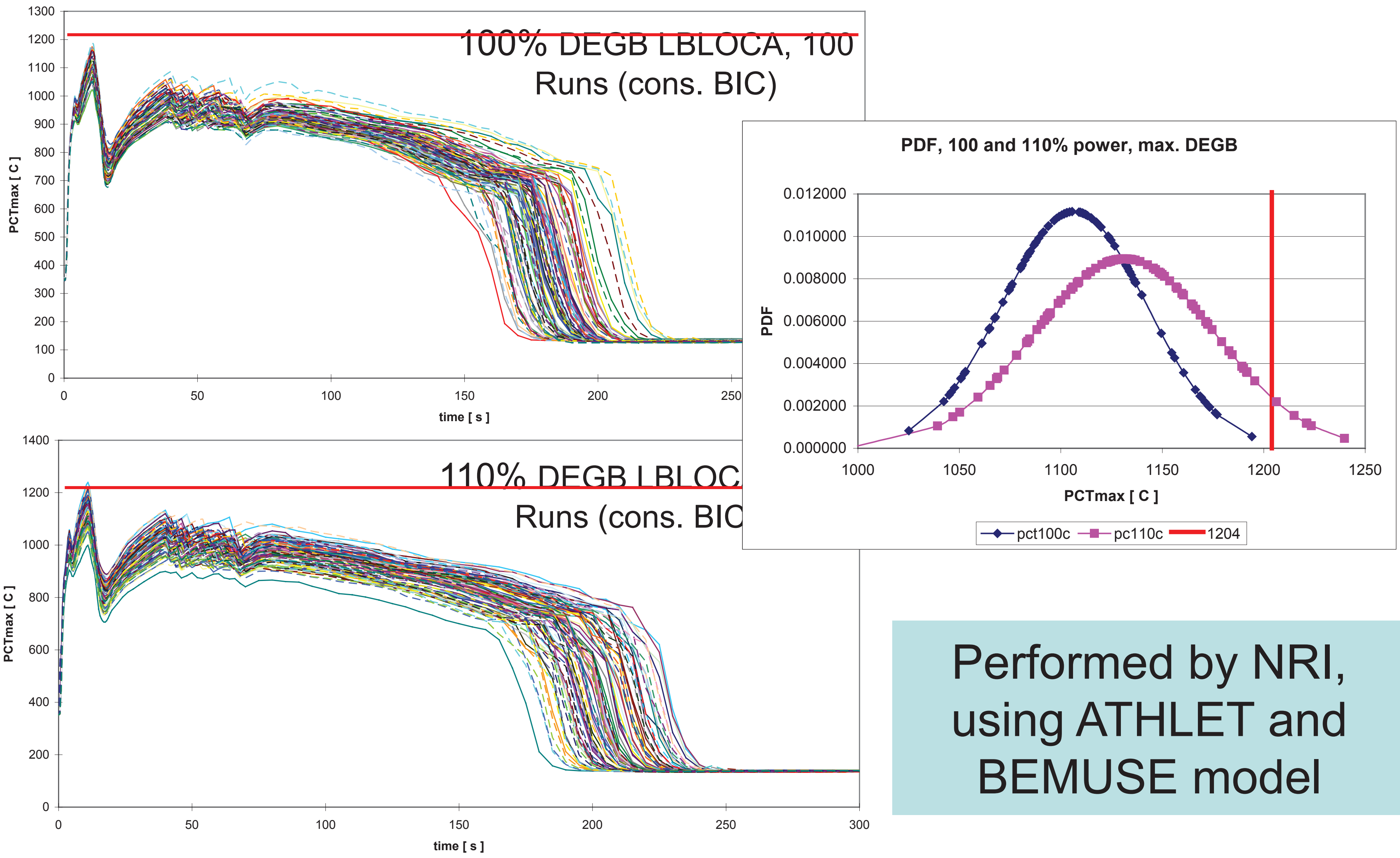
Implementation by SM2A

- Short-lived and focused Appraisal of SMAP methodology required, not real-life application
- Application should reflect multiple changes, including plant changes

➡ Hypothetical (simple) 10% Power Uprate for Zion PWR

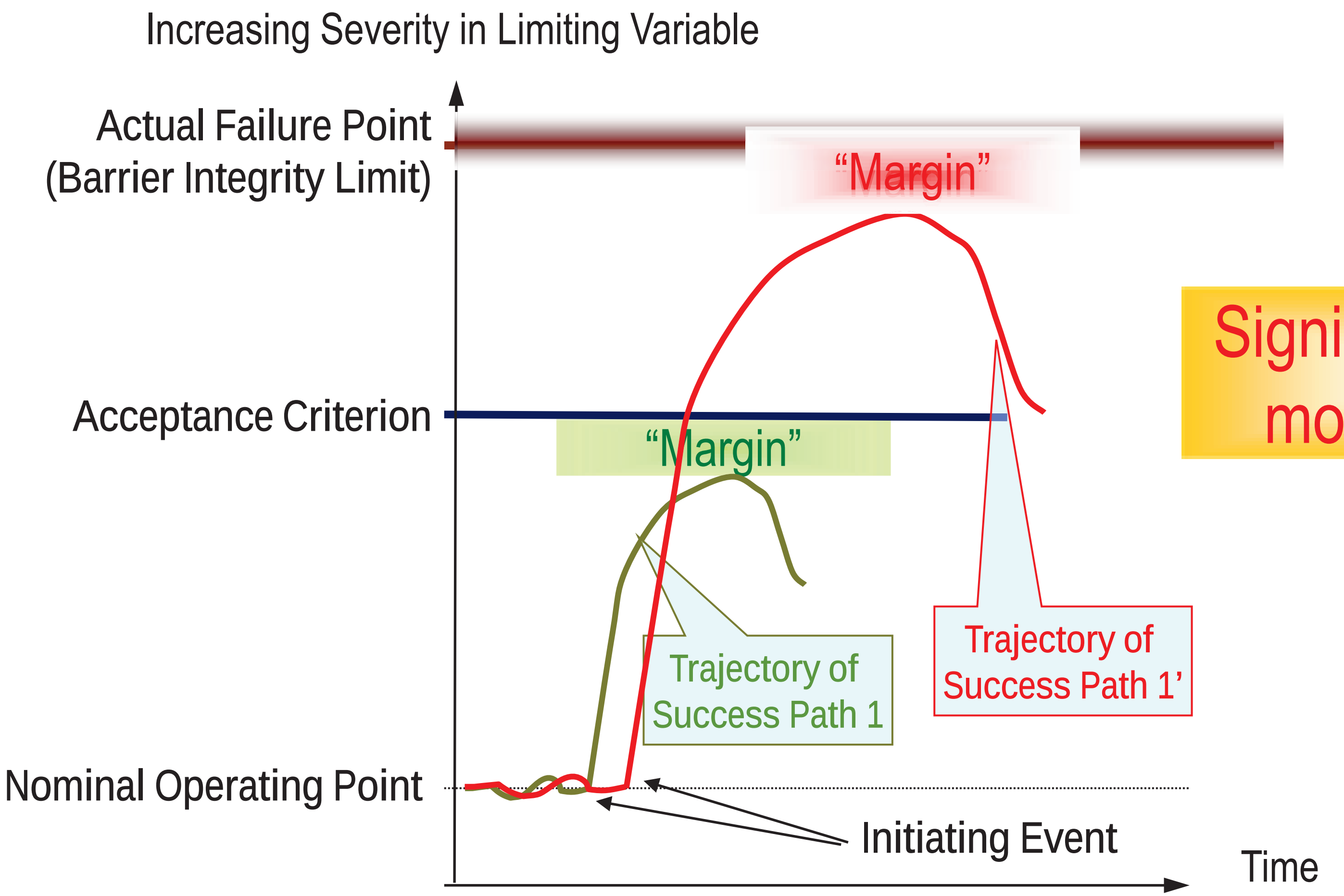
  - Decommissioned plant without sister plants
  - Was studied in NUREG-1150
  - (limited) PRA-documentation available (event trees)
  - Many participants already had input deck (from BEMUSE)

Why 110% power uprate?



Performed by NRI, using ATHLET and BEMUSE model

Where to look?



Increasing Severity in Limiting Variable

Actual Failure Point
(Barrier Integrity Limit)

“Margin”

Acceptance Criterion

“Margin”

Trajectory of
Success Path 1

Trajectory of
Success Path 1’

Nominal Operating Point

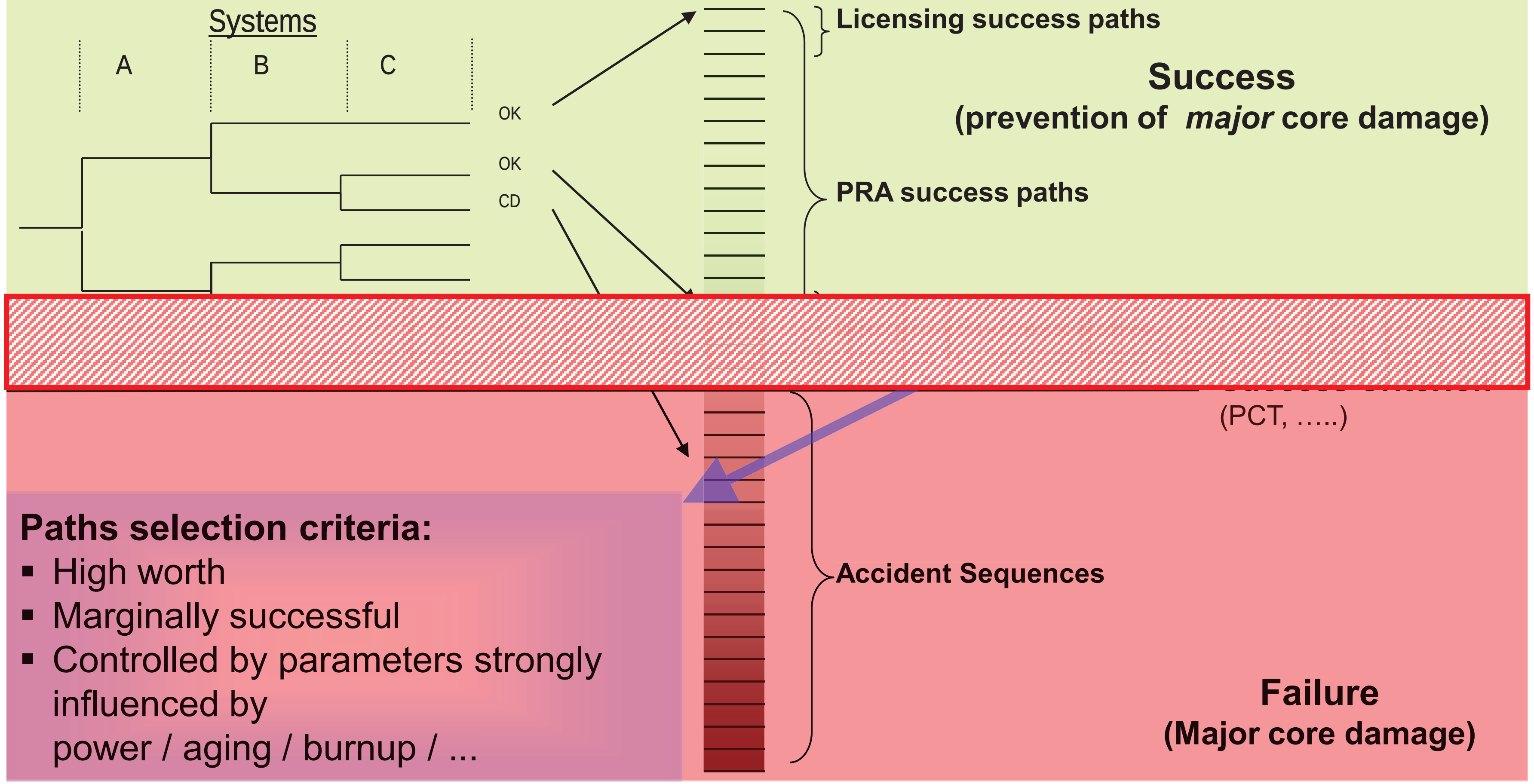
Initiating Event

Time

Significant plant
modification

SM2A: Which are the interesting sequences?

Event Tree Sequence End States, Sorted By “traditional” Margin



Licensing success paths

Success
(prevention of *major* core damage)

PRA success paths

(PCT, …..)

Accident Sequences

Failure
(Major core damage)

Paths selection criteria:

- High worth
- Marginally successful
- Controlled by parameters strongly influenced by power / aging / burnup / ...

Scope of SM2A pilot application

Type	Org	# Seq.	Code	Org
LLOCA	EDF	2	CATHARE/TRACE	EDF/NRI
MLOCA	PSI	1	TRACE	PSI
SLOCA	PSI	1	TRACE	PSI
LOSP	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 +**		

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Summary of SM2A results

Change of CDF	**Scenario**	**Section**	Sequenc Frequency (/r.y)	100% conditional exceedance probability	110% conditional exceedance probability	100% Core Damage Frequency	110% Core Damage Frequency	Delta Core Damage Frequency
Increase > 1E-7/r.y	Loss of Offsite Power + Loss of FW	5.2.1.2	2.7E-5	0.104	0.122	2.81E-6	3.29E-6	4.8E-7
	Loss of Service Water/ Component Cooling Water	5.2.3	1.88E-3			1.34E-6	1.46E-6	1.18E-7
	MBLOCA (3)	5.2.5	9.4E-4	0.123	0.286	8.39E-7	1.64E-6	8.01E-7
Increase << 1E-7/r.y	Loss of Offsite Power + Seal LOCA	5.2.1.1	5.9E-7	-	-	-	-	<<1.E-7 (4)
	Steam Line Break + Loss of FW (1)	5.2.2	2.3E-7 4.35E-11	0.34	0.81			<<1.0E-7
	LBLOCA (2)	5.2.4	4.8E-7			2.49E-9	2.52E-9	2.8E-11
	SBLOCA	5.2.6	1.61E-5	-	-	-	-	<<1.E-7 (4)
No increase. Only a shift of PCT	Turbine Trip	5.2.9	?	0.0	0.0			0.0
No shift of the PCT	Loss of Main Feedwater	5.2.7	1.0E-7	0.0	0.0			0.0
	SGTR	5.2.8	2.793E-7	3.33E-3	3.33E-3	9.30E-10	9.30E-10	0.0

LLOCA: Modifications of ET

Objectives of ET modifications

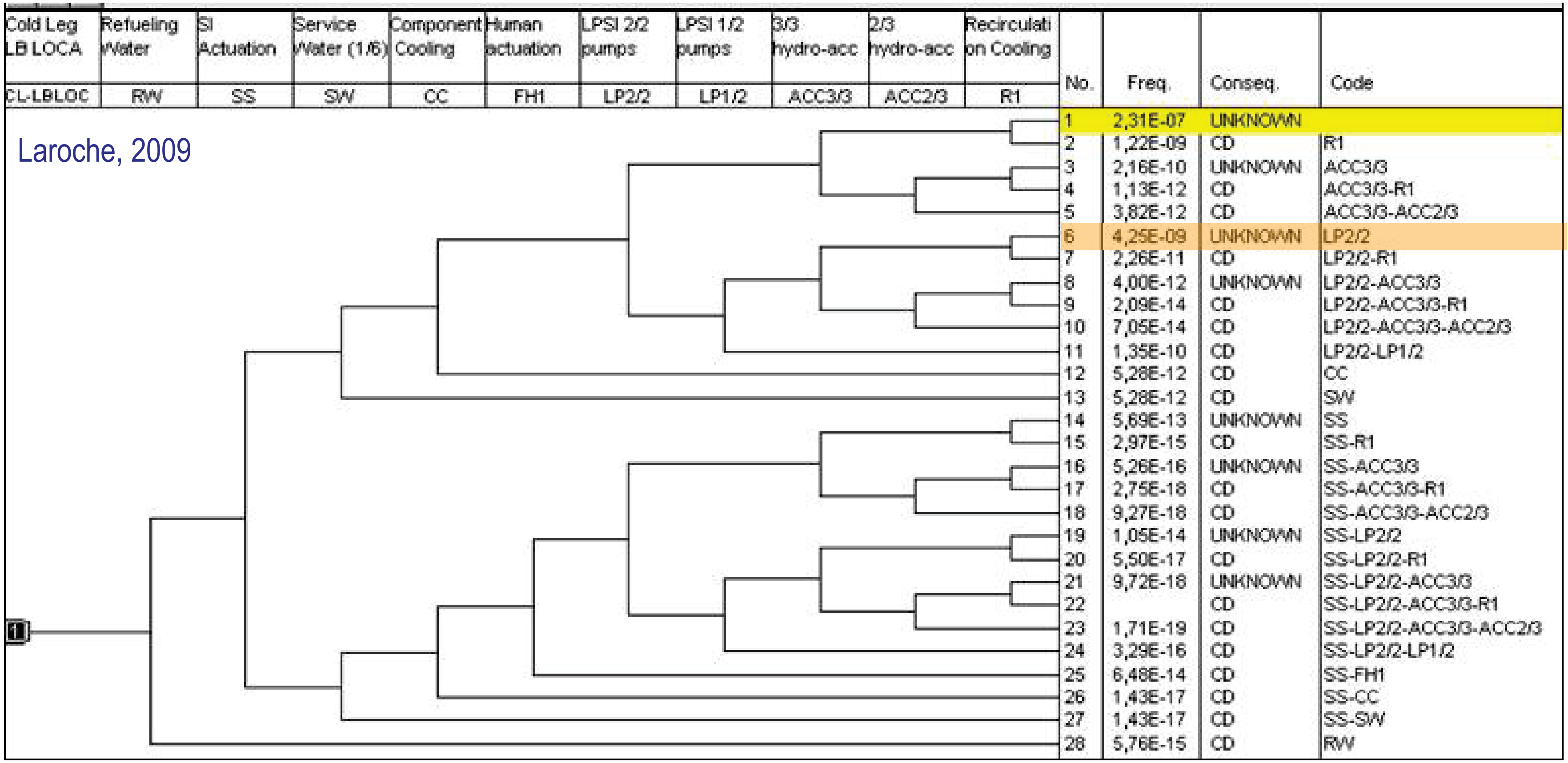
- make ET delineation valid for « before » and « after » status
- make consistent (and realistic) ET delineation and physical scenarios to be calculated
  - each sequence should correspond to a unique non-ambiguous TH scenario so that sequence frequency calculation (PRA) and the consequence calculation (det. transient analysis) are consistent
- to define « frontier » between probabilistic (i.e. PRA model) and deterministic (i.e TH code) models
  - identify input parameters and define adequate probabilistic treatment methodology

LLOCA: Modifications of ET

- LP header « 1/2 LPSI pumps and 3/3 accumulators » assumes 4th accumulator not available and lost to the break
  - Consistent with Cold Leg Break as break may be located on SIT line
  - Overly conservative for Hot Leg break where all 4 SITs may be available to refill lower plenum
- ET-1 has been divided into two ET
  - HL-ET-1 which describes Hot Leg Break
  - CL-ET-1 which describes Cold Leg Break
  - Initiating event frequency (i.e. 9,4E-4/y.r for Run1) has been equally distributed between HL and CL break events (i.e. 4,7E-4/y.r for HL break event and 4,7E-4/y.r for CL break event)

PAUL SCHERRER INSTITUT PSI

LLOCA: Final CL-ET-1

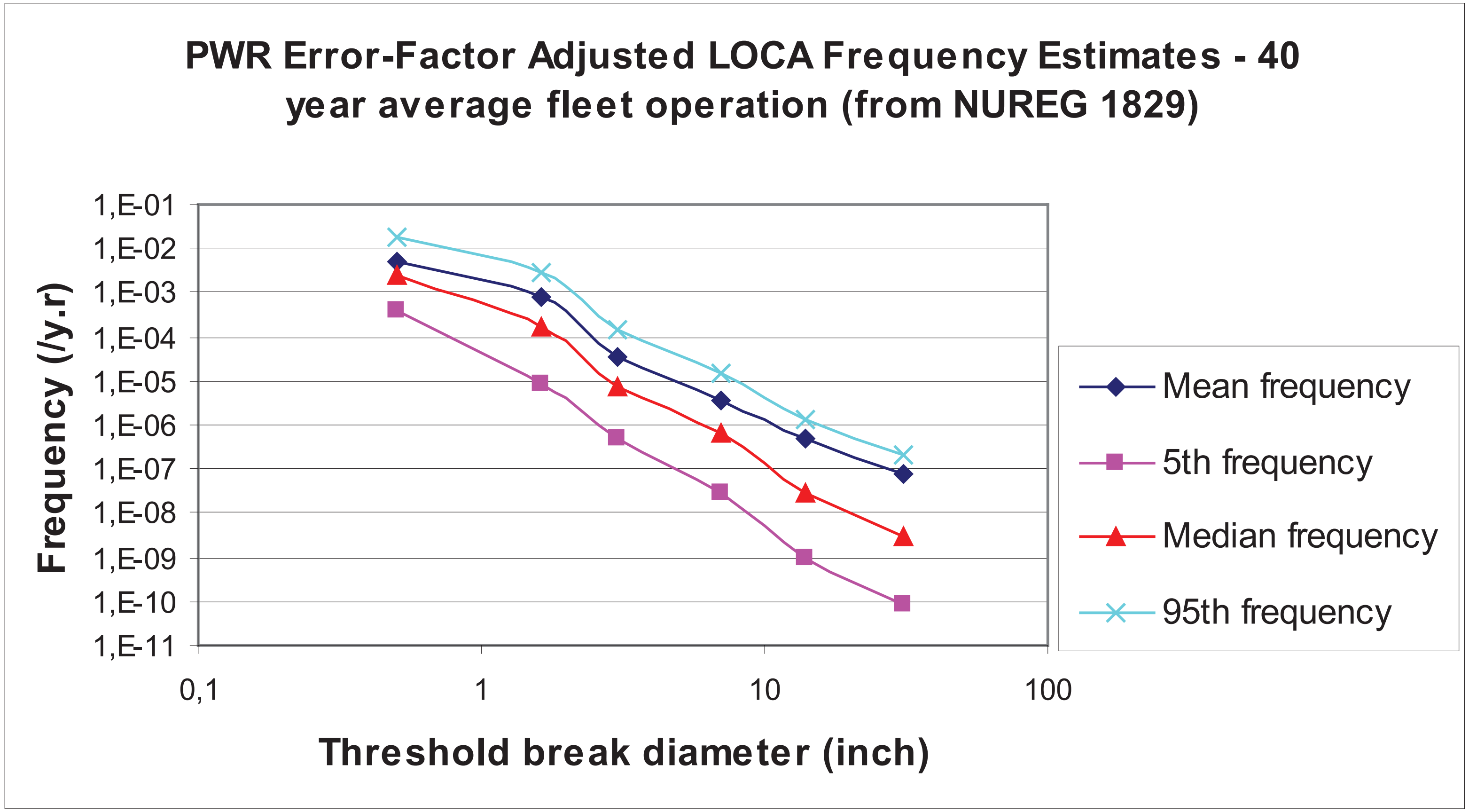


Laroche, 2009

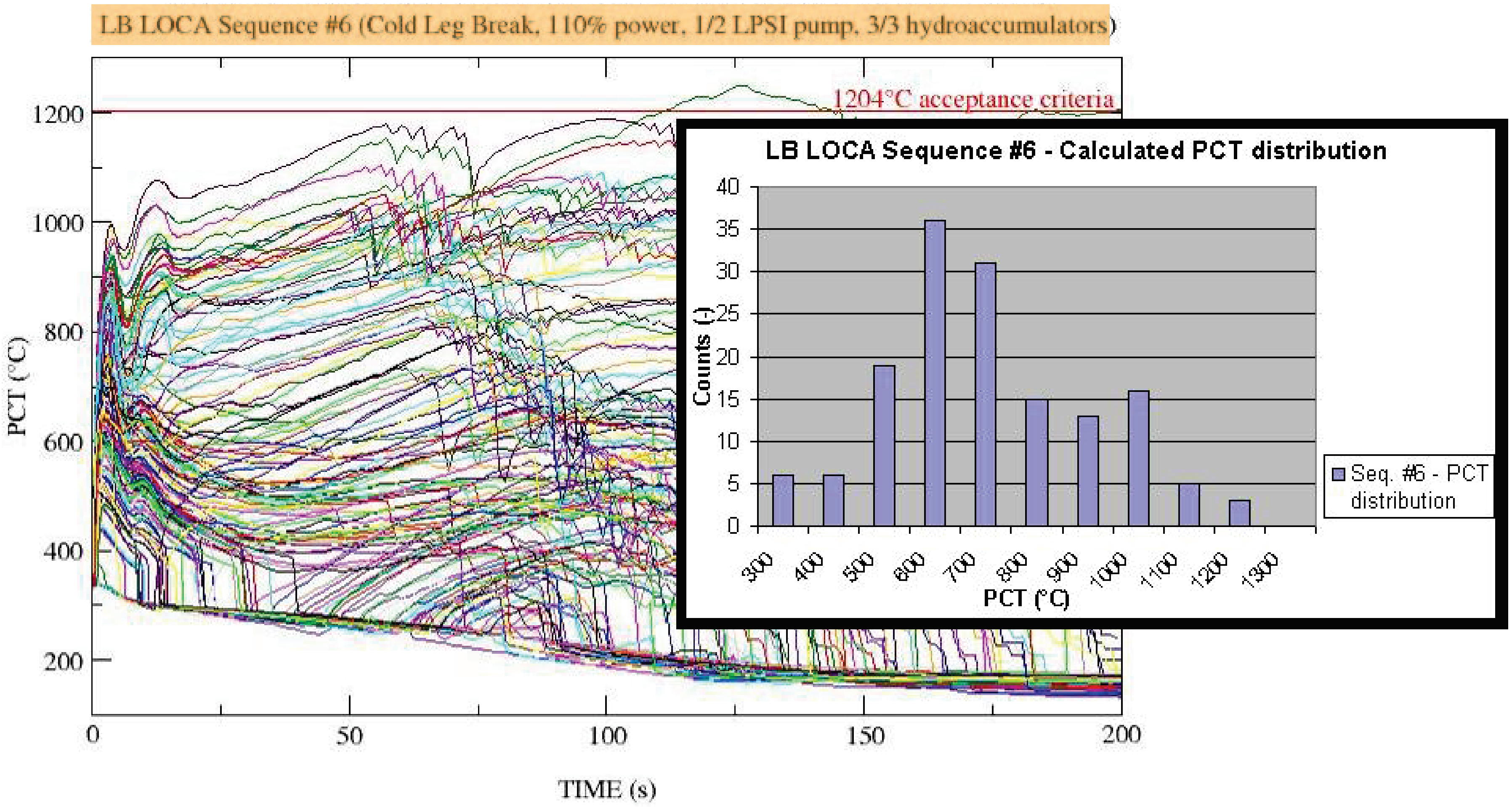
No.	Freq.	Conseq.	Code
1	2,31E-07	UNKNOWN	
2	1,22E-09	CD	R1
3	2,16E-10	UNKNOWN	ACC3/3
4	1,13E-12	CD	ACC3/3-R1
5	3,82E-12	CD	ACC3/3-ACC2/3
6	4,25E-09	UNKNOWN	LP2/2
7	2,26E-11	CD	LP2/2-R1
8	4,00E-12	UNKNOWN	LP2/2-ACC3/3
9	2,09E-14	CD	LP2/2-ACC3/3-R1
10	7,05E-14	CD	LP2/2-ACC3/3-ACC2/3
11	1,35E-10	CD	LP2/2-LP1/2
12	5,28E-12	CD	CC
13	5,28E-12	CD	SW
14	5,69E-13	UNKNOWN	SS
15	2,97E-15	CD	SS-R1
16	5,26E-16	UNKNOWN	SS-ACC3/3
17	2,75E-18	CD	SS-ACC3/3-R1
18	9,27E-18	CD	SS-ACC3/3-ACC2/3
19	1,05E-14	UNKNOWN	SS-LP2/2
20	5,50E-17	CD	SS-LP2/2-R1
21	9,72E-18	UNKNOWN	SS-LP2/2-ACC3/3
22		CD	SS-LP2/2-ACC3/3-R1
23	1,71E-19	CD	SS-LP2/2-ACC3/3-ACC2/3
24	3,29E-16	CD	SS-LP2/2-LP1/2
25	6,48E-14	CD	SS-FH1
26	1,43E-17	CD	SS-CC
27	1,43E-17	CD	SS-SW
28	5,76E-15	CD	RW

- 2/2 LPSI pumps operating+ 3/3 hydro-accumulators available + success of recirculation switching

LLOCA: Break size as probabilistic parameter



LLOCA: Final CL-ET-1


LB LOCA Sequence #6 (Cold Leg Break, 110% power, 1/2 LPSI pump, 3/3 hydroaccumulators)

Negligible ΔCDF evaluated

Summary of SM2A results

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Increase << 1E-7/r.y	Loss of Offsite Power + Seal LOCA	5.2.1.1	5.9E-7	-	-	-	-	<<1.E-7 (4)
	Steam Line Break + Loss of FW (1)	5.2.2	2.3E-7 4.35E-11	0.34	0.81			<<1.0E-7
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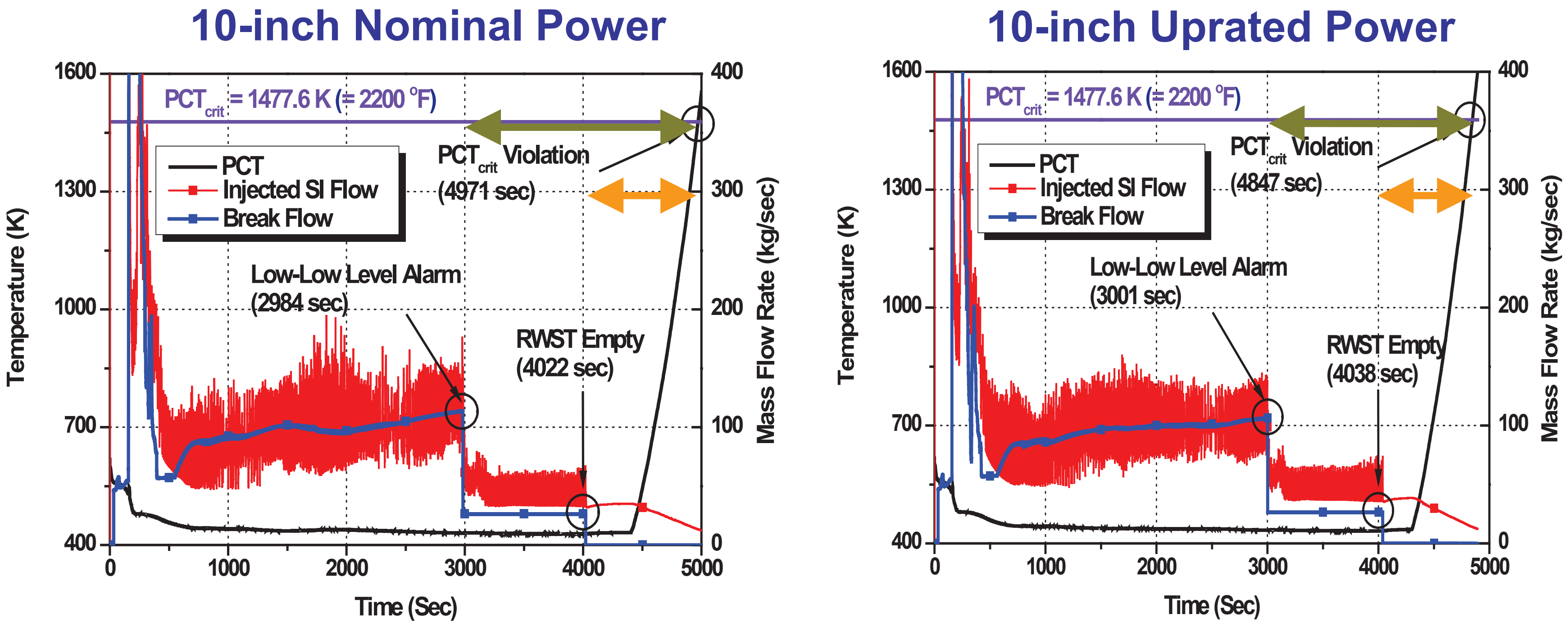
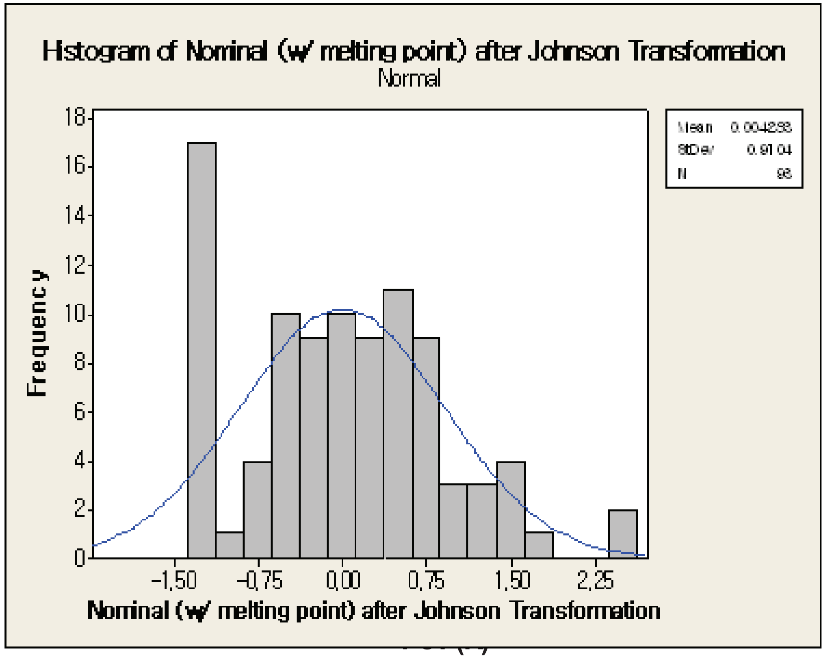
	SBLOCA	5.2.6	1.61E-5	-	-	-	-	<<1.E-7 (4)
No increase. Only a shift of PCT	Turbine Trip	5.2.9	?	0.0	0.0			0.0
No shift of the PCT	Loss of Main Feedwater	5.2.7	1.0E-7	0.0	0.0			0.0
	SGTR	5.2.8	2.793E-7	3.33E-3	3.33E-3	9.30E-10	9.30E-10	0.0

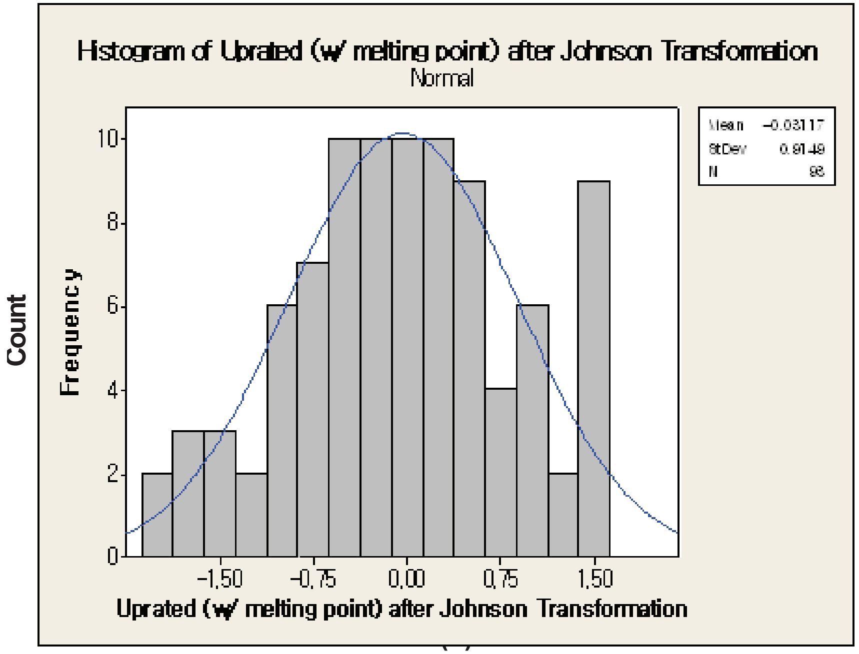
MLOCA: Transient results



- Time from low-low level alarm to critical PCT for nominal and uprated power
  - 1987 sec (~33.1 min) and 1846 sec (~30.8 min), respectively
- Time from complete depletion of RWST to critical PCT
  - 949 sec (~15.8 min) and 809 sec (~13.5 min), respectively
- Times from low-low level alarm to RWST empty are identical in both cases

MLOCA: PDFs





$$P(T > 1477.6K) = P(Z_{\text{nominal}} > 1.0606) = 0.1230$$

$$P(T > 1477.6K) = P(Z_{\text{uprated}} > 0.4870) = 0.2856$$

$$Z_{nom} = 0.915005 + 0.586868 * \ln\left(\frac{T - 579.450}{2178.15 - T}\right)$$

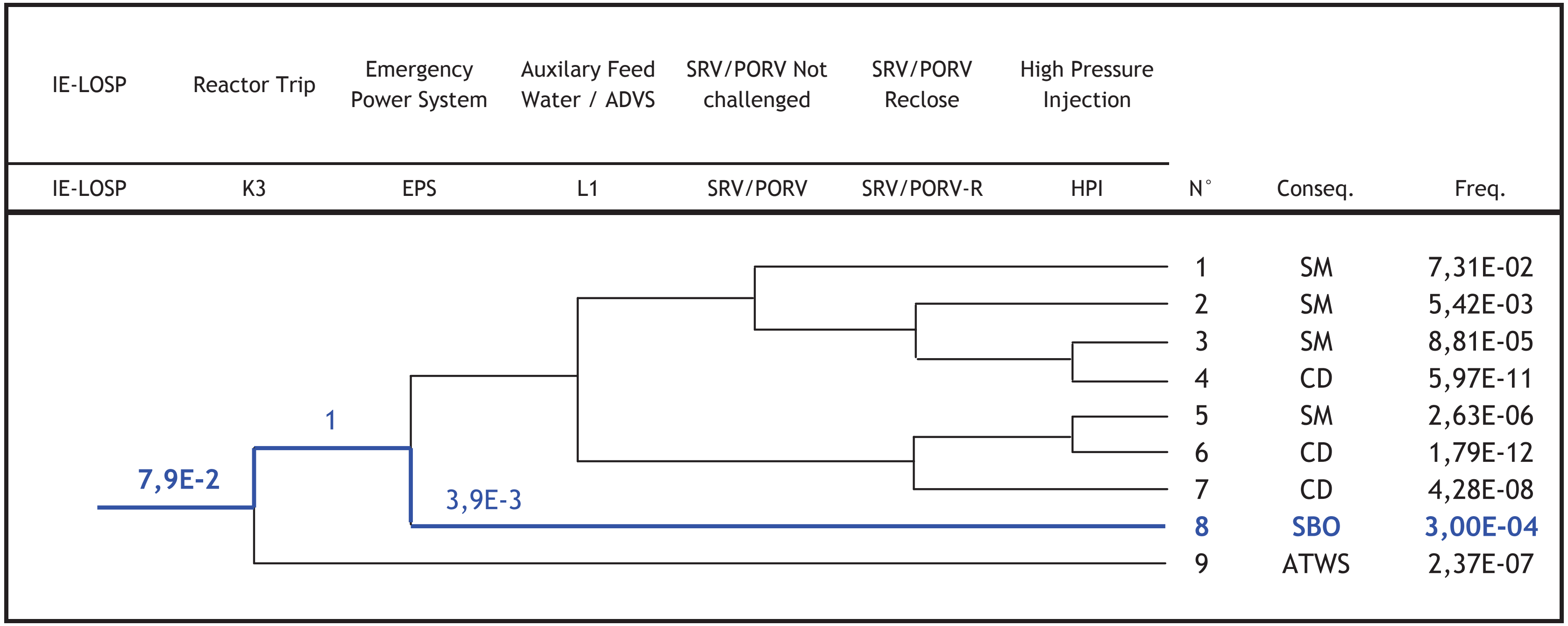
$$Z_{uprated} = 0.740128 + 0.740313 * \ln\left(\frac{T - 715.779}{2549.92 - T}\right)$$

Exceedance probability increases from 0.123 to 0.286
(Δ= 0.163) due to power uprate of 10 %

Summary of SM2A results

Change of CDF	**Scenario**	**Section**	Sequenc Frequency (/r.y)	100% conditional exceedance probability	110% conditional exceedance probability	100% Core Damage Frequency	110% Core Damage Frequency	Delta Core Damage Frequency
Increase > 1E-7/r.y	Loss of Offsite Power + Loss of FW	5.2.1.2	2.7E-5	0.104	0.122	2.81E-6	3.29E-6	4.8E-7
	Loss of Service Water/ Component Cooling Water	5.2.3	1.88E-3			1.34E-6	1.46E-6	1.18E-7
	MBLOCA (3)	5.2.5	9.4E-4	0.123	0.286	8.39E-7	1.64E-6	8.01E-7
Increase << 1E-7/r.y	Loss of Offsite Power + Seal LOCA	5.2.1.1	5.9E-7	-	-	-	-	<<1.E-7 (4)
	Steam Line Break + Loss of FW (1)	5.2.2	2.3E-7 4.35E-11	0.34	0.81			<<1.0E-7
	LBLOCA (2)	5.2.4	4.8E-7			2.49E-9	2.52E-9	2.8E-11
	SBLOCA	5.2.6	1.61E-5	-	-	-	-	<<1.E-7 (4)
No increase. Only a shift of PCT	Turbine Trip	5.2.9	?	0.0	0.0			0.0
No shift of the PCT	Loss of Main Feedwater	5.2.7	1.0E-7	0.0	0.0			0.0
	SGTR	5.2.8	2.793E-7	3.33E-3	3.33E-3	9.30E-10	9.30E-10	0.0

LOSP: Event tree



IE-LOSP	Reactor Trip	Emergency Power System	Auxilary Feed Water / ADVS	SRV/PORV Not challenged	SRV/PORV Reclose	High Pressure Injection
IE-LOSP	K3	EPS	L1	SRV/PORV	SRV/PORV-R	HPI

N°	Conseq.	Freq.
1	SM	7,31E-02
2	SM	5,42E-03
3	SM	8,81E-05
4	CD	5,97E-11
5	SM	2,63E-06
6	CD	1,79E-12
7	CD	4,28E-08
8	SBO	3,00E-04
9	ATWS	2,37E-07

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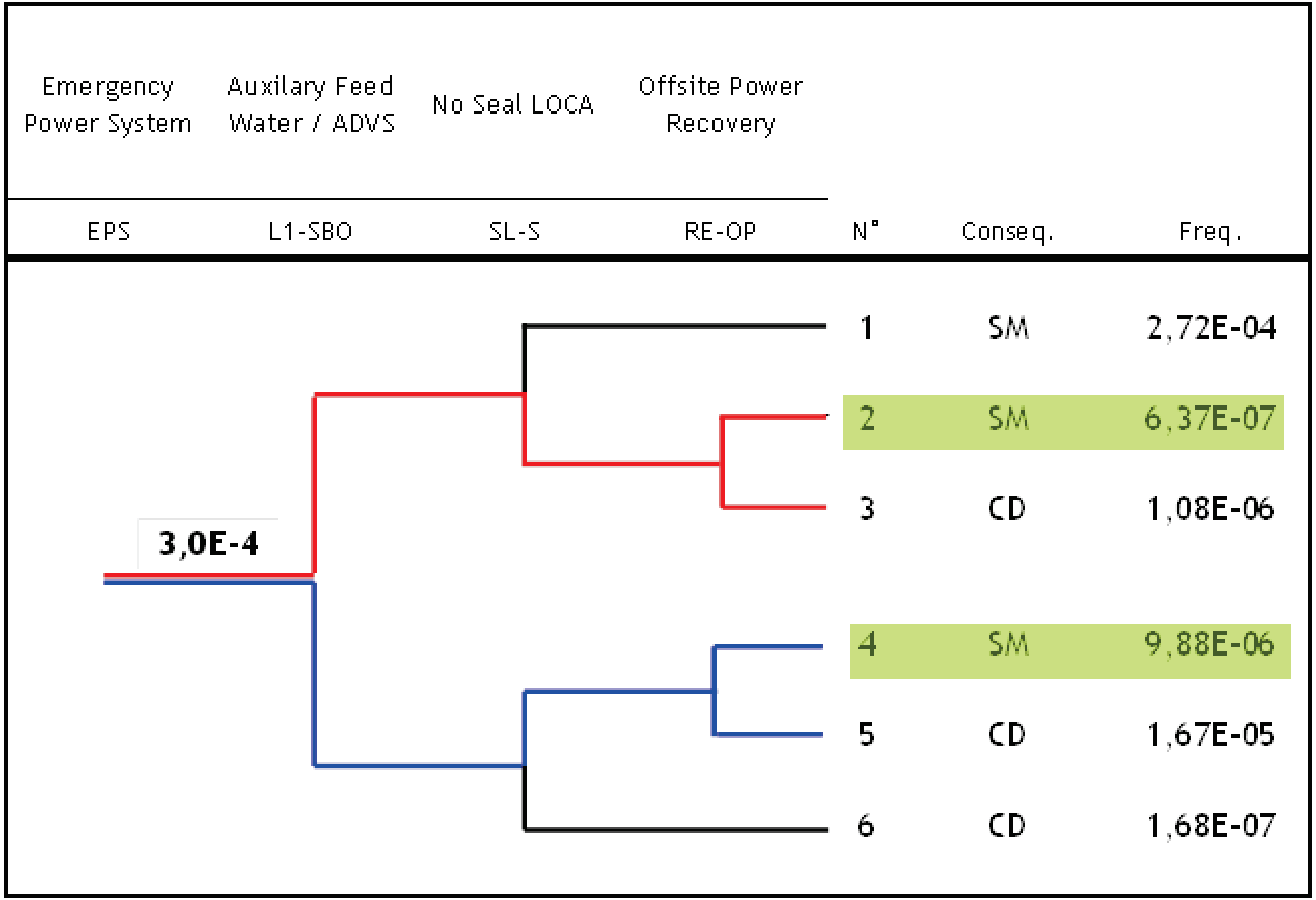
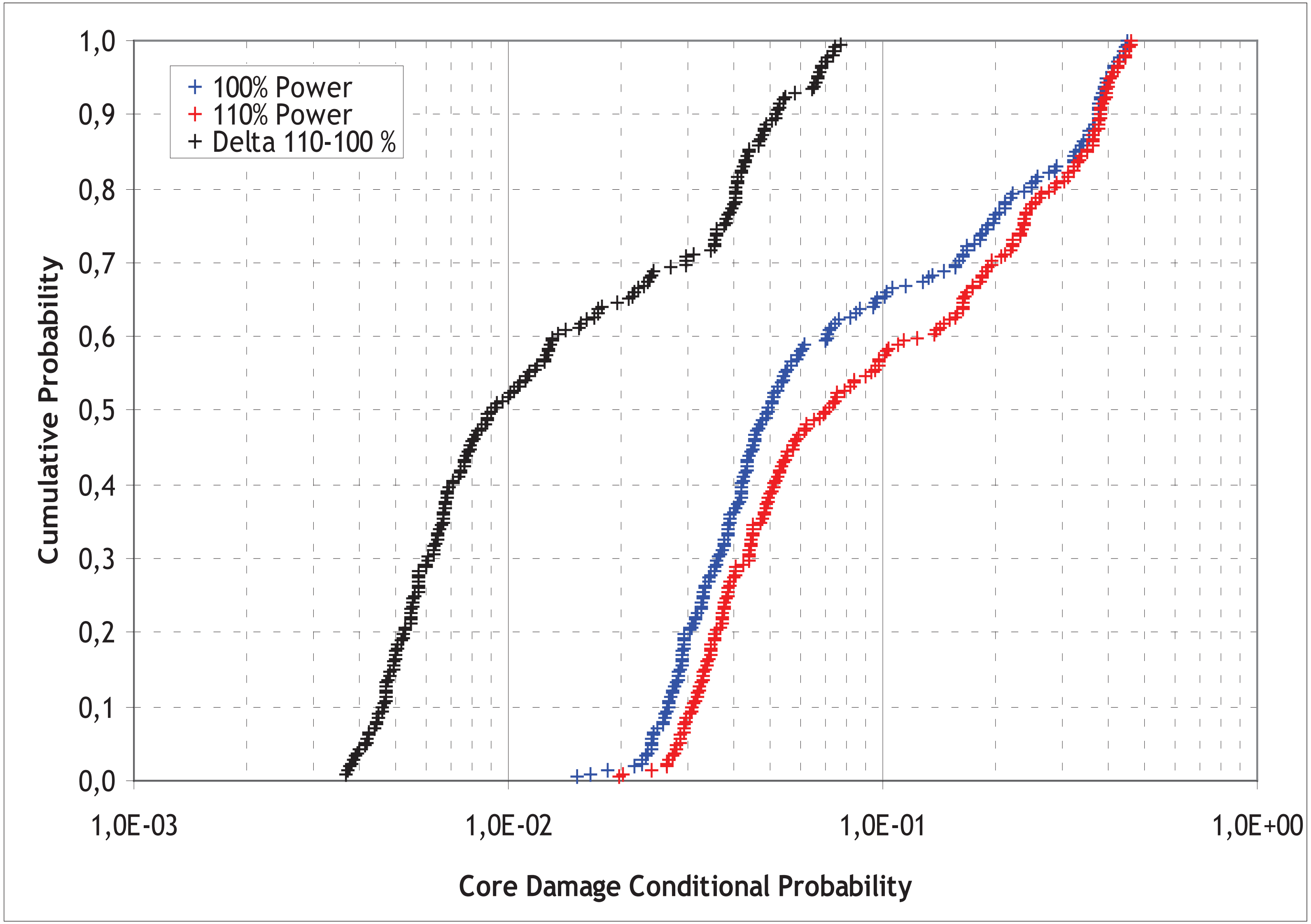
LOSP +Loss of AFW: Event trees



Emergency Power System	Auxilary Feed Water / ADVS	No Seal LOCA	Offsite Power Recovery			
EPS	L1-SBO	SL-S	RE-OP	N°	Conseq.	Freq.
				1	SM	2,72E-04
				2	SM	6,37E-07
				3	CD	1,08E-06
				4	SM	9,88E-06
				5	CD	1,67E-05
				6	CD	1,68E-07

LOSP: Results for SBO + Loss of AFW



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Achievements of SM2A

- SMAP **framework** was proven **workable** for evaluation of safety margins.
  - Following **refinements** were needed
    - Screening of PSA sequences
    - Reformulation of existing event trees
- **Increase of probability of exceedance** for surrogate limit indicating core damage was **successfully evaluated** for chosen scenarios from several ET's
  - Impact of power uprate could also be traced for scenarios with no criteria violation!
- Approach makes best **benefit from already validated simulation models**!

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Achievements of SM2A (II)

- First success in building a team of analysts understanding both probabilistic and deterministic safety analysis
- First large-scale BEPU campaign!
- Final report published summer 2011
  - Should become available soon at http://www.oecd-nea.org/nsd/docs/indexcsni.html

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Lessons learned

- Very close interaction between transient simulation experts and probabilistic analysis experts key to successful application of SMAP framework
- Efficient PSA sequence screening is very important
- Selection of time delay distributions (operator actions) becomes quite crucial.
- SM2A calculation would not be expected to yield a significant contributor to ΔCDF

Lessons learned (contd)

- Combination of PRA techniques with transient simulation requires “discrete” system configurations
- Introduction of “TH” header for the conditional probability of exceedance into the different sequences can be useful for a straight-forward aggregation of the frequency of exceedance

Limitations of SM2A study

- Restricted to PCT as single safety parameter (PCT)
- Limited scope of events: No ATWS, ISLOCA, RIA, Scenarios occurring at Low Power, External Events
- Some modeling assumptions had to be made in light of minimal resources available
  - Assumed simplified power uprate procedure
  - Restriction to 0D-kinetics
  - No containment modeling
- Only “simple” PSA available

Limitations of SM2A (II)

- Different partners use different T/H codes: possible heterogeneity in the results
  - Input model qualification?
- Sequence screening might have been too simple / coarse:
  - Cut-off limit: 10^{-7}/yr / 10^{-6}/yr
- Accuracy of exceedance frequency evaluation might be affected by availability of limited computing resources (-> # runs)

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Stimulating New BEPU Approaches?

- Well validated codes and models should be applied
  - Range of “validated” individual code application might well be surpassed
- Consideration of several barriers and safety variables requires adopting different codes, e.g. TRACE – COBRA, TRACE - MELCOR.
  - efficient BEPU-procedure for chained codes (propagation of uncertainty from code to code)
- Development of computational tools should allow for automating the analysis to a high degree.
  - Discrete dynamic event tree (DDET) tools to address the automatic identification of sequence branching points and respective sequence simulations.
  - Combination with BEPU may pose new challenges:
    - transition from one sequence to another one may occur due to perturbation of uncertain parameters, e.g. by reaching trip levels.
    - adequate number of simulation cases (as stipulated by the Wilks formula) allocated to individual sequences must be ensured.
    - requires reconsidering the sampling strategy (biased sampling, GA, …)
  - How to ensure the quality of the calculations in large-scale deployments?
    - Automated post-processing assessing quality
    - Clustering of results?

Stimulating New BEPU Approaches?

- Evaluation of scenarios of long duration could benefit from employing a set of plant (input) models with graded levels of modeling detail for different phases of the transient:
  - set of models staged in modeling detail
  - would help to reduce computational effort,
  - would reduce amount of generated data.
  - Time step size Restrictions could possibly be lessened
- Collection of (evaluated) experimental data into publicly available data bases (as e.g. the criticality handbook of NSC) appears as an important desideratum if not a necessity.
  - Good starts are seen in the SET and IET collections of CSNI CCVM databases,
  - new OECD projects all are providing modern data sets that after a protection period become publically available.
- Engineering judgment, ideally combined with a systematic expert solicitation process, could fill the gaps in uncertainty information where experimental evidence is lacking.
  - Likely, it will be an important source of uncertainty characterization especially for phenomena occurring during low-frequency sequences.

Thank you for your attention!

Questions?

International Atomic Energy Agency

Combining Insights from Probabilistic and Deterministic Safety Analyses in Option 4 from the IAEA Specific Safety Guide SSG-2

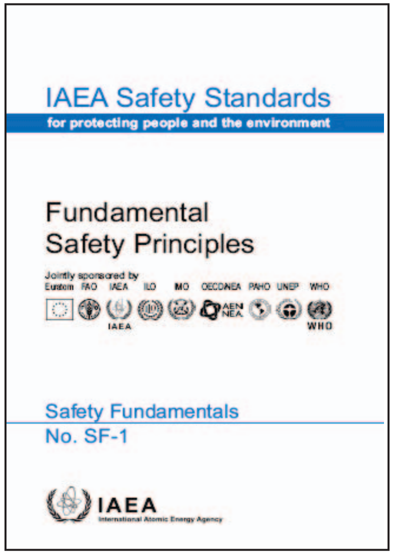
Milorad Dusic; Mark Dutton; Horst Glaeser; Joachim Herb; Javier Hortal; Rafael Mendizabal; Fernando Pelayo

OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations
Barcelona, Spain
16 – 18 November 2011

Safety Assessment in the IAEA Safety Standards

SAFETY FUNDAMENTALS

Principles 5 and 8



IAEA Safety Standards
for protecting people and the environment

Fundamental Safety Principles

Jointly sponsored by
Euratom FAO IAEA ILO IMO OECD/NEA PAHO UNEP WHO

Safety Fundamentals
No. SF-1

IAEA
International Atomic Energy Agency

2006

SAFETY REQUIREMENTS

Comprehensive & detailed analysis required

IAEA SAFETY STANDARDS SERIES

Safety of Nuclear Power Plants: Design

REQUIREMENTS

No. NS-R-1

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA

2000
2011

SAFETY REQUIREMENTS:
Safety Assessment for Facilities and Activities
GS-R-4

2008

SAFETY GUIDES

New SGs provide guidance on performing DSA and PSA and their application

Deterministic Safety Analysis for NPPs
SSG-2

Development & Application of Level 1 PSA for NPPs
SSG-3

Development & Application of Level 2 PSA for NPPs
SSG-4

2008/09

Specific Safety Guide SSG-2

- Objective and Scope
  - To provide harmonized guidance to designers, operators, RBs and TSOs with respect to:
    - Performing deterministic safety analysis and,
    - For utilizing the results of such analysis for safety and reliability improvements in nuclear power plants
  - Applicable to current and future designs

Content of the Safety Guide SSG-2

1. Introduction
2. Nuclear Power Plant States and Conditions Classifications
3. Deterministic Safety Analysis and Acceptance Criteria
4. Conservative Deterministic Safety Analysis
5. Best Estimate (BE) Plus Uncertainty Analysis
6. Verification and Validation of Computer Codes
7. Deterministic Safety Analysis in Relation to Engineering Aspects of Safety and Probabilistic Safety Analysis
8. Application of Deterministic Safety Analysis
9. Source Term Evaluation for Normal and Accidental Operating States

4 OPTIONS from 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

Load and Barrier Probability Distributions (Fig 1)

- **Distribution of code predictions/results** is a consequence of uncertainties in I&B conditions data as well as in computer model
- **Distribution of failures** i.e. values where the barrier fails is a consequence of our limited knowledge of the precise phenomenon that causes failure

Figure 1: Load and Barrier Probability Distributions

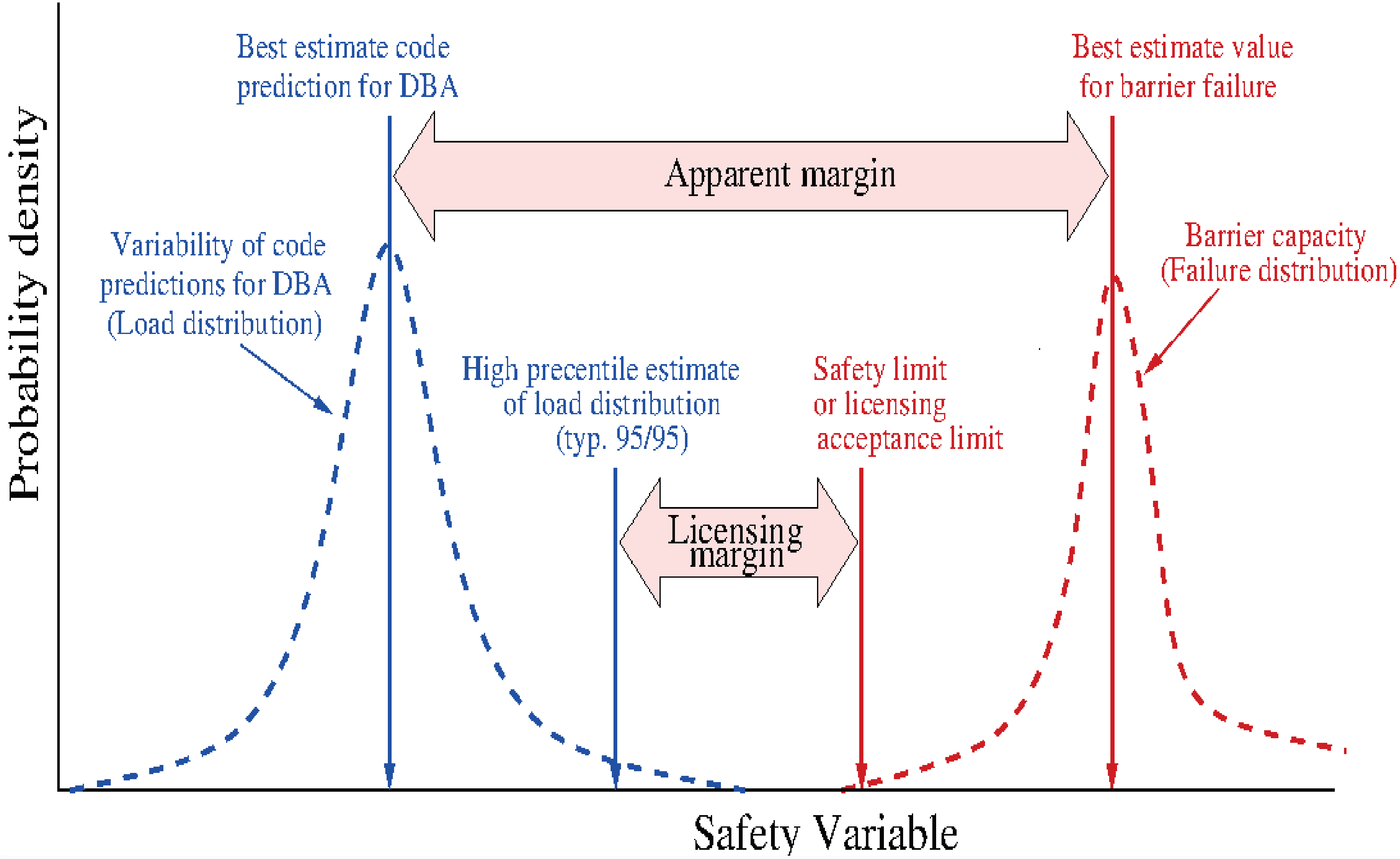
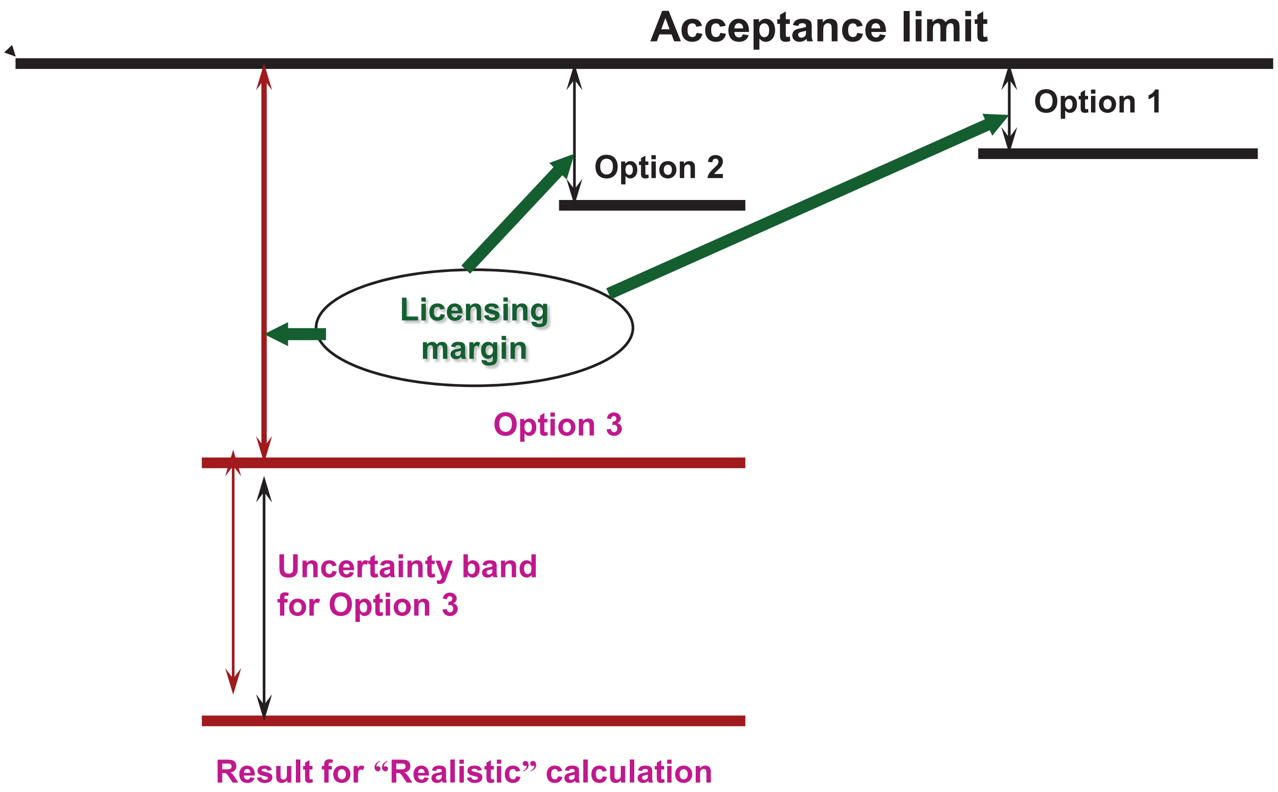
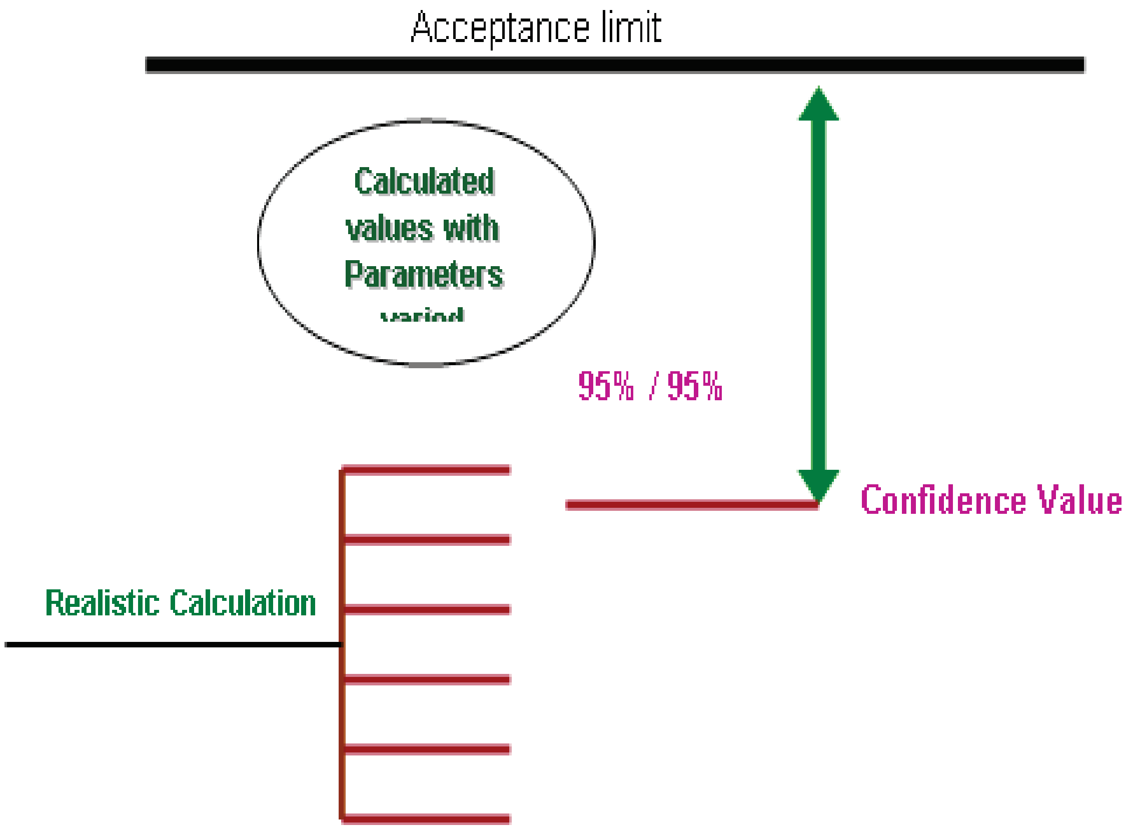


Fig 2: Licensing Margins under Options 1, 2 and 3



International Atomic Energy Agency

Fig. 3 Illustrative example of Licensing Margin for Option 3



Acceptance limit

Calculated values with Parameters varied

95% / 95%

Confidence Value

Realistic Calculation

Availability of Safety Systems

- Common feature for options 1, 2 and 3 is the conservative assumption on the availability of trains of safety systems – typically in Option 3 we assume one train/pump available.
- The purpose of Option 4 is to provide a further level of flexibility by using probabilistic arguments to take credit for the finite probability that a train is available that was excluded deterministically when Option 3 was employed.
- Another feature in this option is that we relate the acceptance limit to the frequency of the sequence following each Postulated Initiating Event (PIE).

Fig. 4: Dose Limit for Whole Body at Site Boundary

American National Standard ANSI/ANS-51.1-1983

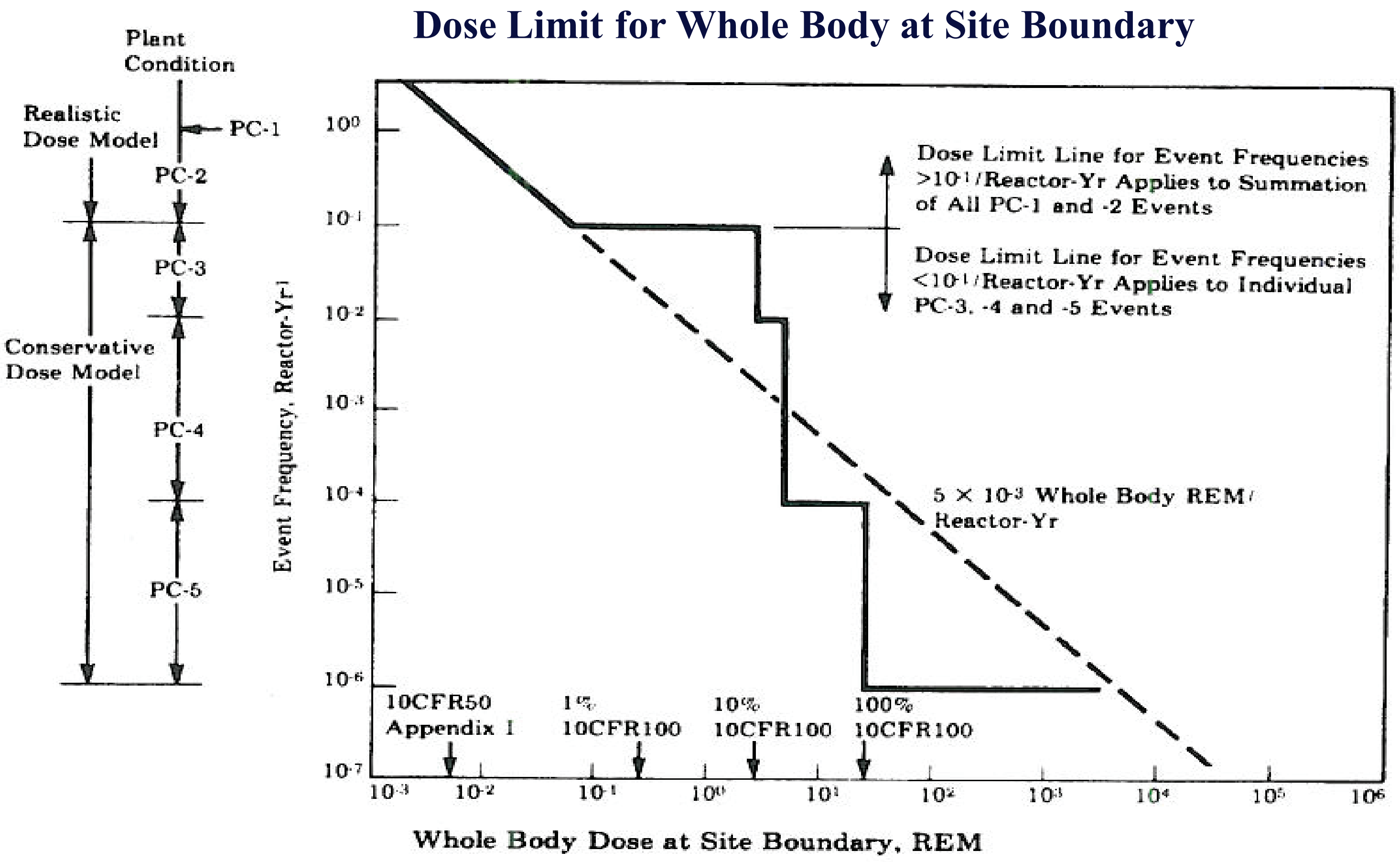
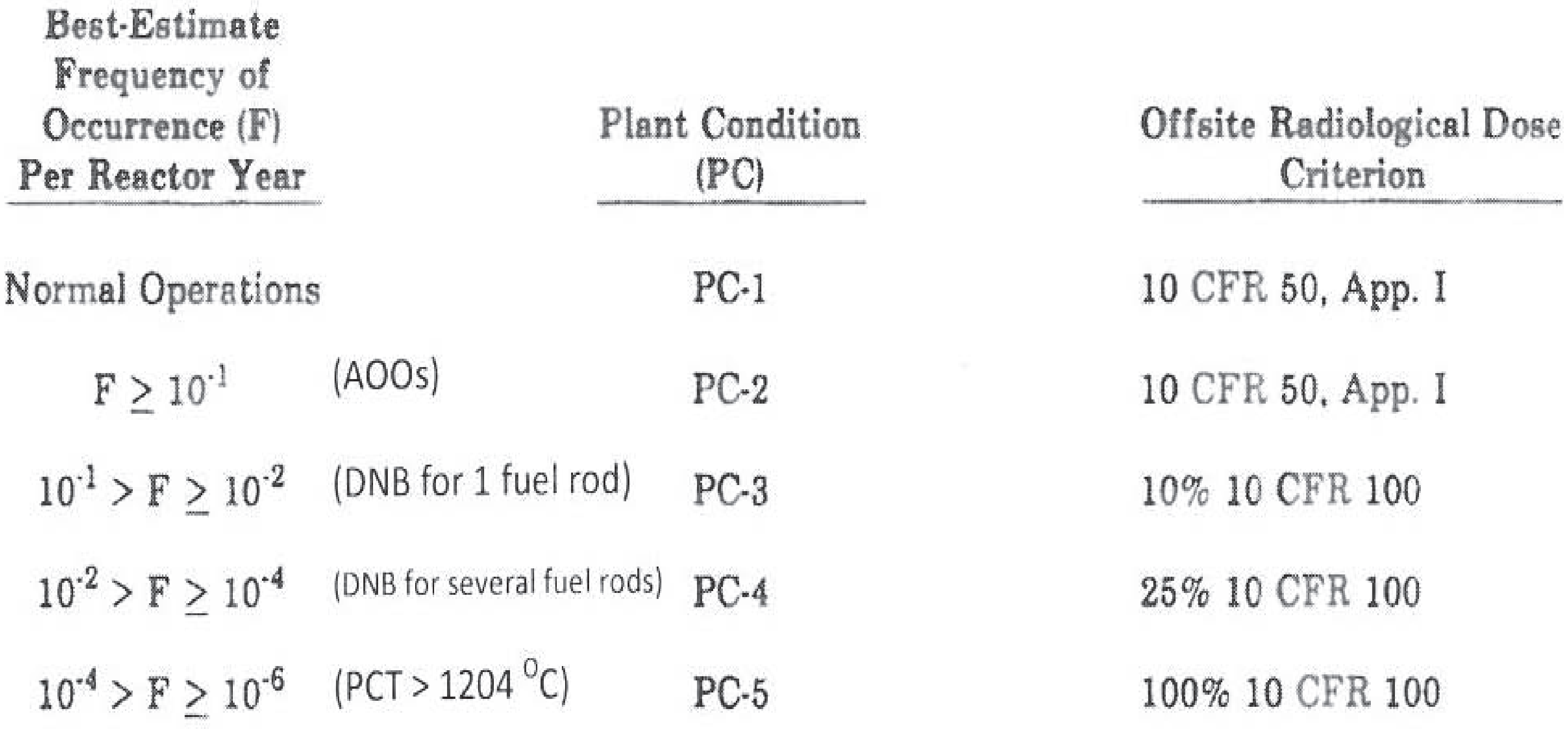


Fig 4 cont.: Offsite Radiological Dose Criteria for Plant Conditions



Best-Estimate Frequency of Occurrence (F) Per Reactor Year		Plant Condition (PC)	Offsite Radiological Dose Criterion
Normal Operations		PC-1	10 CFR 50, App. I
$F \geq 10^{-1}$	(AOOs)	PC-2	10 CFR 50, App. I
$10^{-1} > F \geq 10^{-2}$	(DNB for 1 fuel rod)	PC-3	10% 10 CFR 100
$10^{-2} > F \geq 10^{-4}$	(DNB for several fuel rods)	PC-4	25% 10 CFR 100
$10^{-4} > F \geq 10^{-6}$	(PCT > 1204 °C)	PC-5	100% 10 CFR 100

Option 4

- **Moving from Option 2 to Option 3 we replaced conservative I&B condition data with realistic but uncertain values**
- **But we require 95% uncertainties with 95% confidence to be below the acceptance limit**
- **5% probability of limit exceedance may include single failure sequences**

Option 4

- **Following this reasoning, moving from Option 3 to Option 4 would be to include the availability of safety systems into the uncertainty analysis**
- **Using Option 4 would allow relaxing the single failure criterion for sequences with a very low probabilities of occurrence**
- **On the other hand would provide the possibility of including multiple failures with high likelihood of occurrence**
- **In addition, by using Option 4 we impose additional requirements for those 5% sequences that go beyond the acceptance limit - with this we try to assure that consequences of such sequences are limited and that we do not have any cliff-edge effects**

Option 4

- **After PIE we allow additional failures in safety systems with probabilities $0 < P < 1$**
- **This results in new <u>sequences</u> starting from the PIE (event tree)**
- **Frequency of each sequence is the product of the frequency of the PIE and the probabilities of additional failures**
- **A sequence does not correspond to a single simulation**
- **Within each sequence the uncertainty analysis is performed by choosing different uncertain parameters**
- **A particular choice of parameters is a single simulation and can be termed a <u>simulation run</u>**

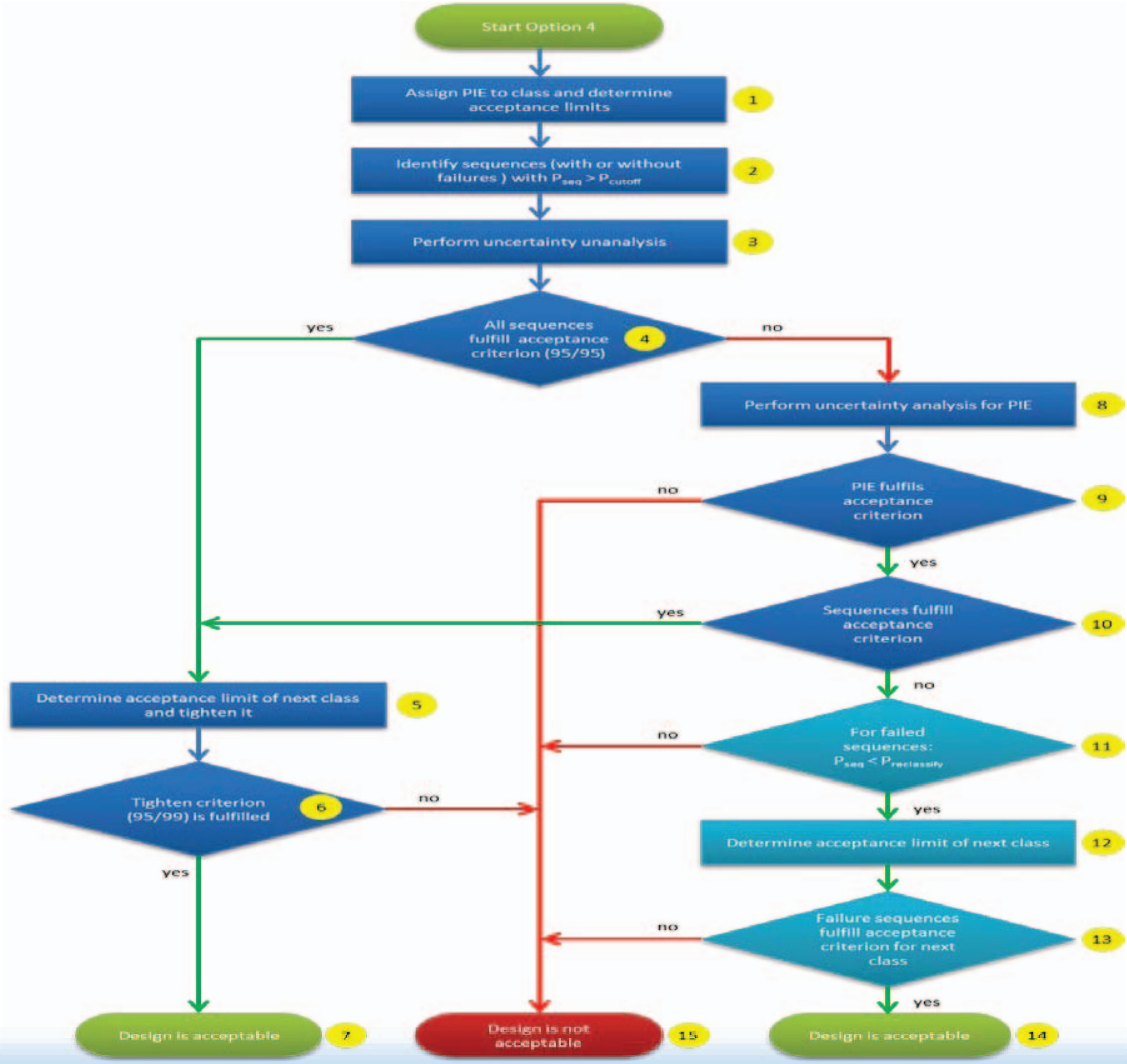
Flow Chart of Option 4 – Left side

- **1. PIE is classified in appropriate accident category and the acceptance limit for that category is identified**
- **2. Identify all sequences starting from PIE and calculate conditional probabilities for each sequence and select $p_{seq} > p_{cutoff}$**
- **3. Perform uncertainty analyses for each sequence (sim. runs)**
- **4. If all sequences fulfill the acc. criteria 95/95 we still impose additional requirements for remaining 5% of simulation runs**
- **5. Determine acceptance limit for the next class and tighten it**
- **6. Demonstrate with a high degree of confidence (95%) for those 5% that 99% of uncertainty results remain below the tightened acc. limit**
- **7. If this is true, design is acceptable**

Flow Chart of Option 4 – Right side

- 5. If some sequences (with additional failures) violate 95/95 criterion
- 8. First perform uncertainty analyses for ALL SEQUENCES TOGETHER for that particular PIE
- 9. Compare with the acceptance criterion 95/95
- 10. If the answer is yes, analyses of individual sequences is performed and compared with the acceptance criteria
- 11. For those sequences that fail to fulfill 95/95 criterion we compare their conditional probability with $p_{reclassify}$
- 12. Determine the acceptance limit for the next class
- 13. 14. 15. If the sequence fulfills the acceptance criteria for the new class, design is acceptable, otherwise not

Option 4 Flow diagram



Start Option 4

Assign PIE to class and determine acceptance limits 1

Identify sequences (with or without failures) with $P_{seq} > P_{cutoff}$ 2

Perform uncertainty unanalysis 3

All sequences fulfill acceptance criterion (95/95) 4

yes

no

Perform uncertainty analysis for PIE 8

PIE fulfils acceptance criterion 9

no

yes

Sequences fulfill acceptance criterion 10

yes

no

For failed sequences: $P_{seq} < P_{reclassify}$ 11

no

yes

Determine acceptance limit of next class 12

Failure sequences fulfill acceptance criterion for next class 13

no

yes

Design is acceptable 14

Determine acceptance limit of next class and tighten it 5

Tighten criterion (95/99) is fulfilled 6

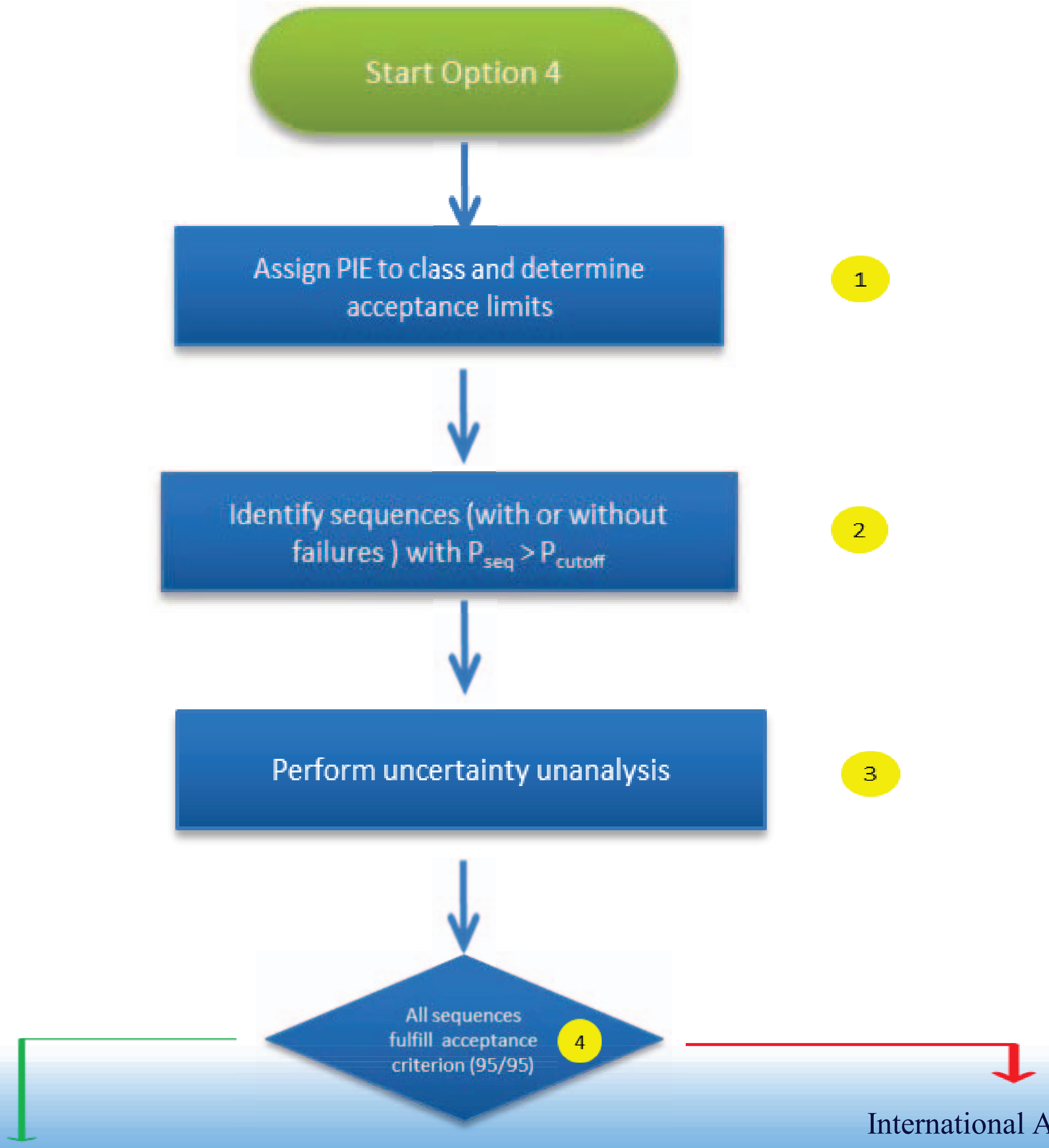
no

yes

Design is acceptable 7

Design is not acceptable 15

Option 4 Flow diagram – top part



Start Option 4

Assign PIE to class and determine acceptance limits

1

Identify sequences (with or without failures) with $P_{seq} > P_{cutoff}$

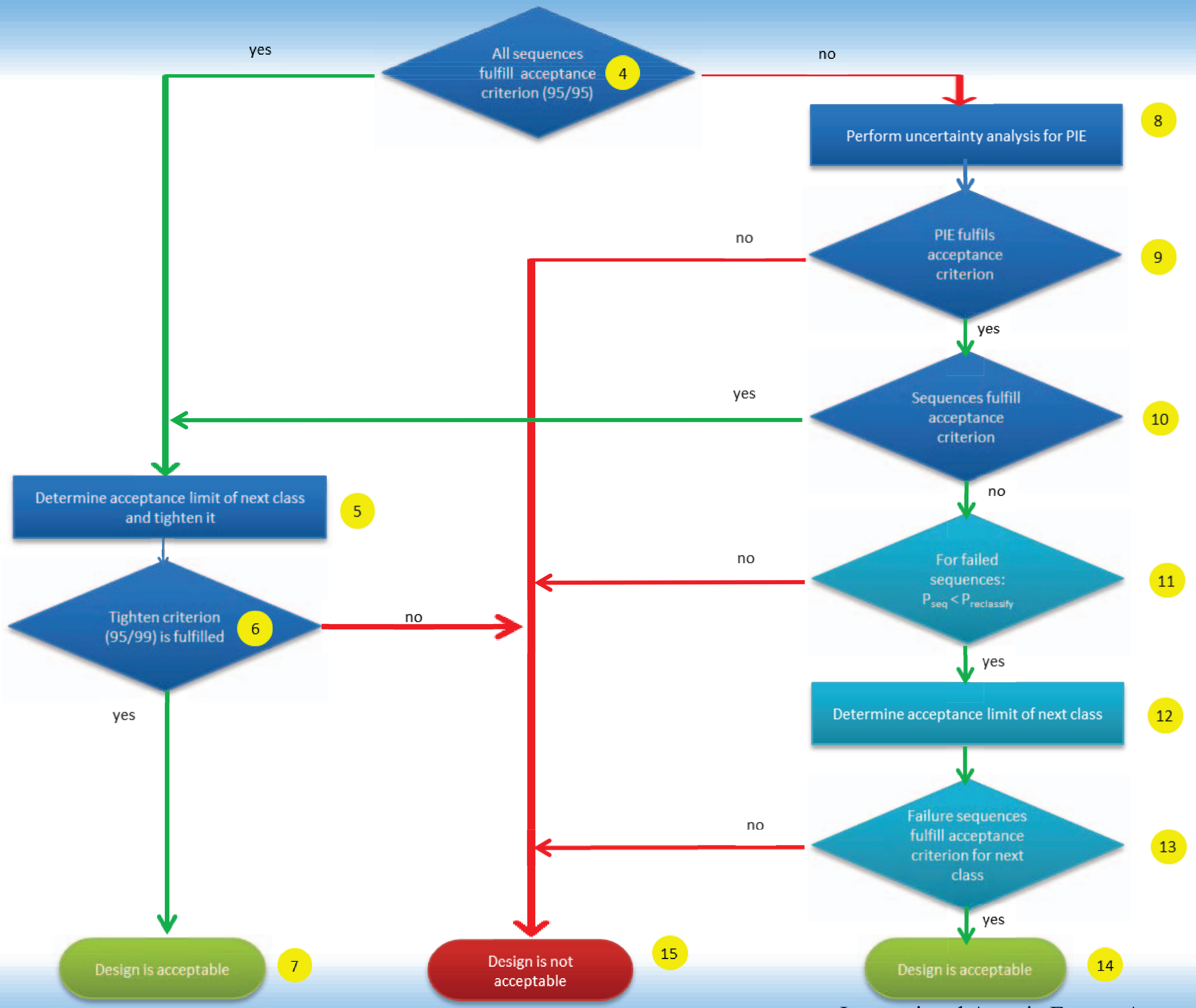
2

Perform uncertainty unanalysis

3

All sequences fulfill acceptance criterion (95/95)

4



All sequences fulfill acceptance criterion (95/95) 4

yes

no

Perform uncertainty analysis for PIE 8

PIE fulfils acceptance criterion 9

no

yes

Sequences fulfill acceptance criterion 10

yes

no

Determine acceptance limit of next class and tighten it 5

Tighten criterion (95/99) is fulfilled 6

no

yes

For failed sequences: $P_{seq} < P_{reclassify}$ 11

no

yes

Determine acceptance limit of next class 12

Failure sequences fulfill acceptance criterion for next class 13

no

yes

Design is acceptable 7

Design is not acceptable 15

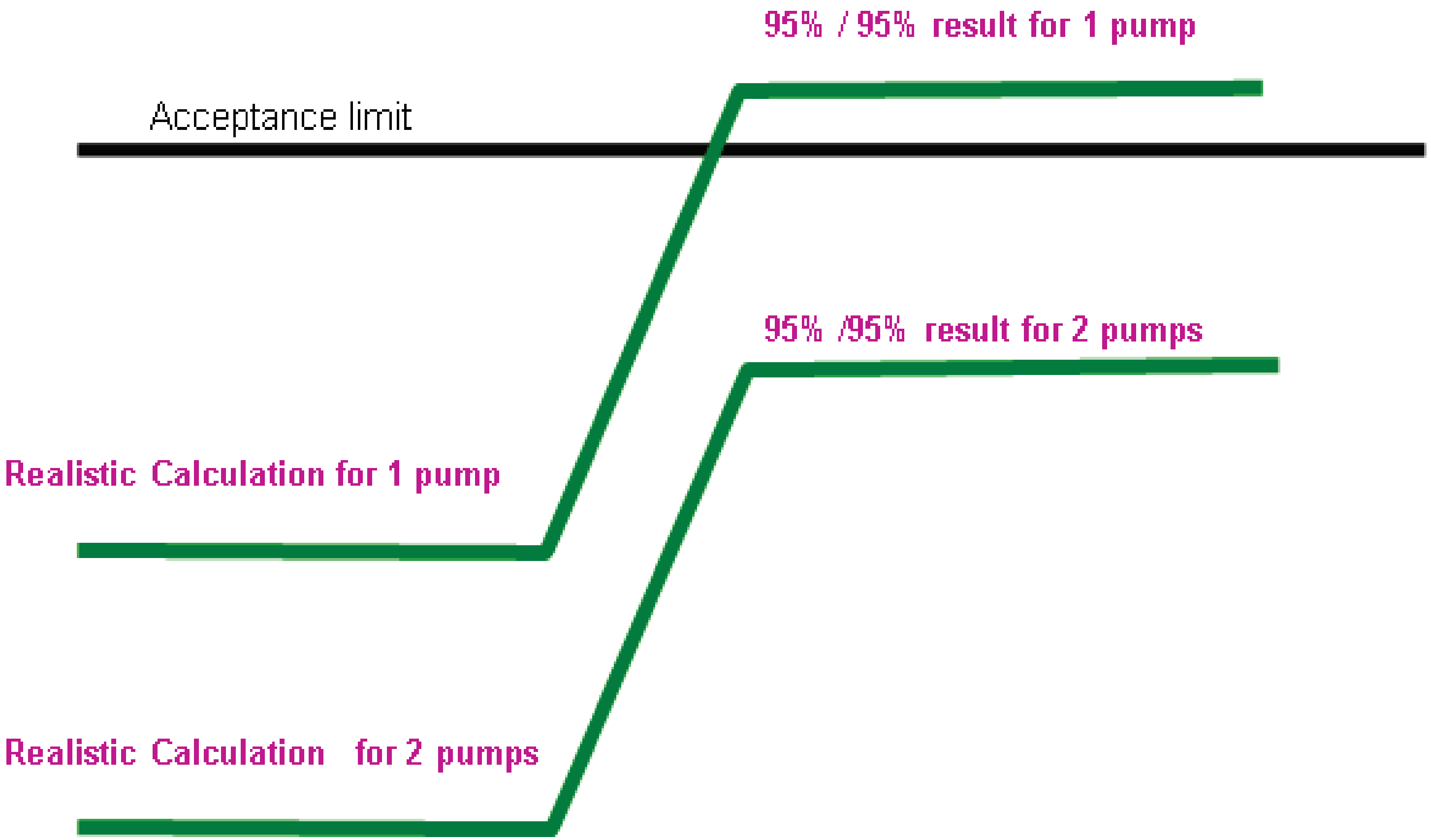
Design is acceptable 14

Example of Option 4

- Case of plant modification (power uprate).
- In the original plant design, before modification Option 3 was used and demonstrated that even with one pump available, acceptance limit is met.
- After plant modification we find that the safety system with only one pump available exceeds the acceptance limit when using Option 3.

Example of Option 4 - cont.

Fig. 6: Acceptance limit exceeded with one pump



95% / 95% result for 1 pump

Acceptance limit

95% /95% result for 2 pumps

Realistic Calculation for 1 pump

Realistic Calculation for 2 pumps

Example of Option 4 - cont.

- If we relate this to the dose diagram as in Fig. 7 we moved from X1 to X2 and exceeded the acceptance limit.
- In Option 3 we have assumed to have only one train available, the second one was considered to be in maintenance and the third one out of service to satisfy the single failure criterion.
- Availability of one train is therefore set to 1 and the availability of two trains to 0.

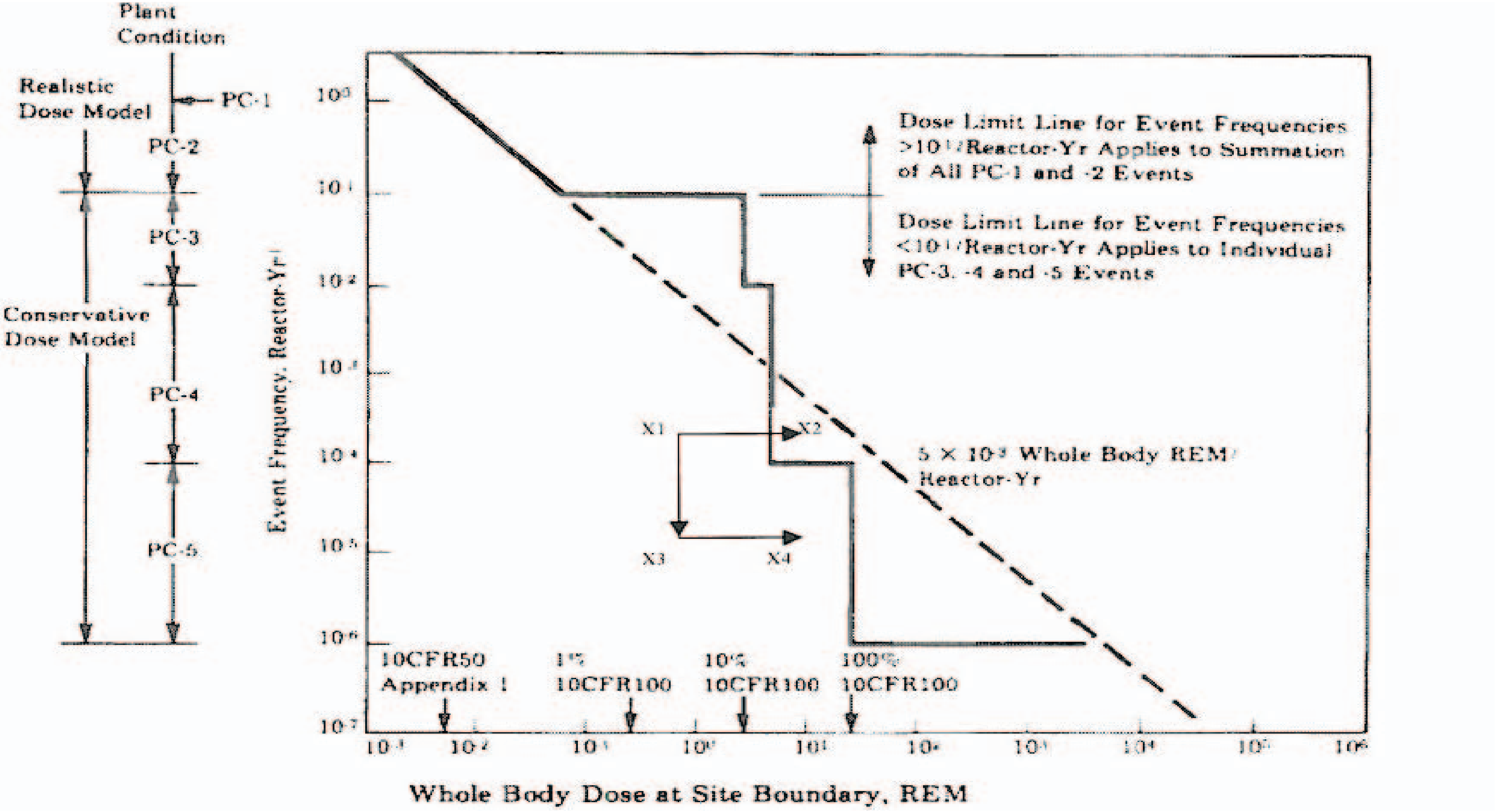
Example of Option 4 – cont.

- **In Option 4 however the probability of having two trains available is calculated on probabilistic grounds and is $0 < P < 1$.**
- **When multiplying the frequency of the transient with the unavailability of anyone of two trains the frequency of the sequence moves down to lower frequencies from X1 to X3 (in Fig 7 to PC-5 range).**
- **After modification X3 moves to point X4 which in this case is below the acceptance limit.**

Example of Option 4 – cont.

Fig. 7: Illustration of an acceptable case where an acceptance limit is exceeded

Dose Limit for Whole Body at Site Boundary



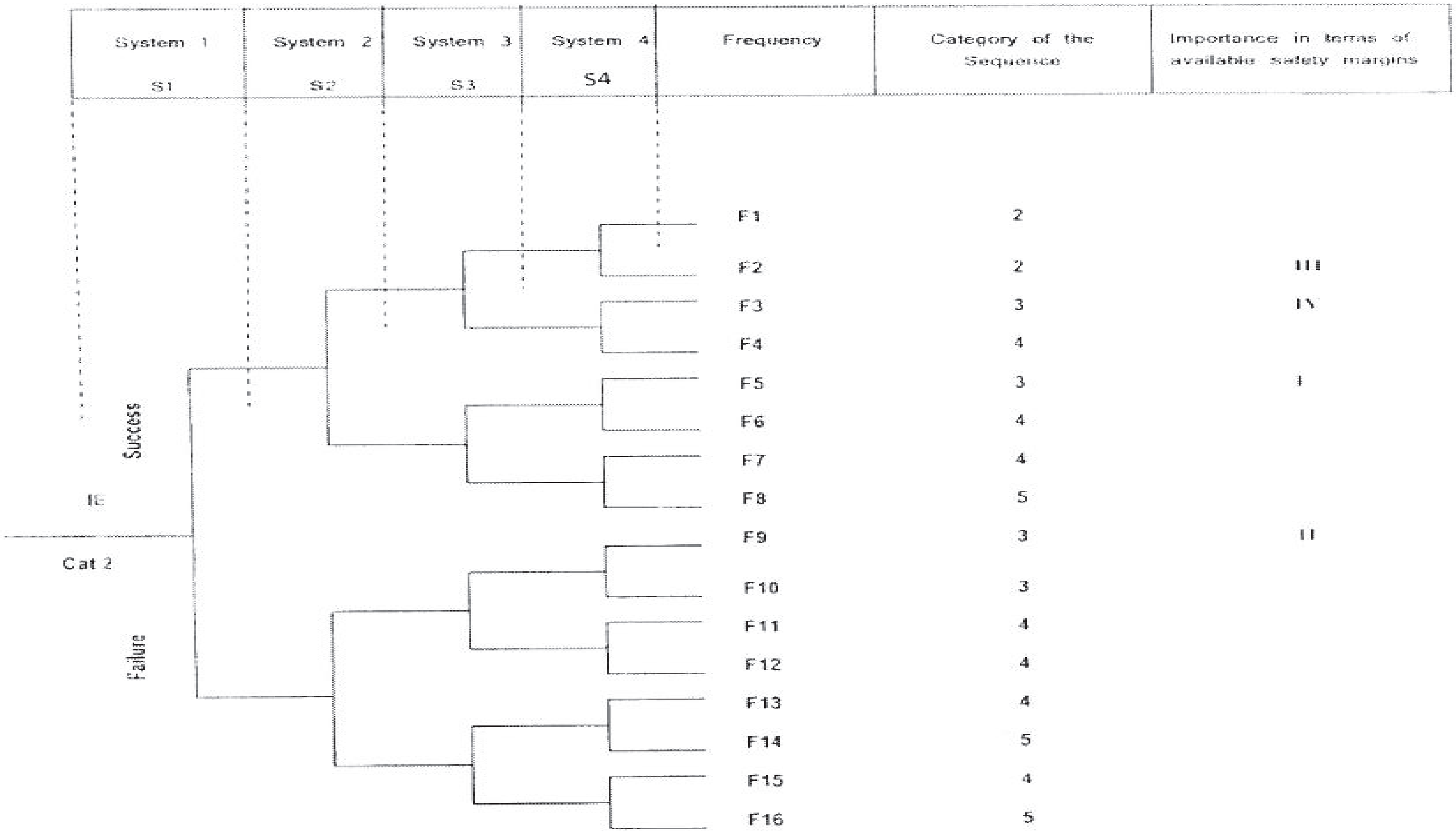
Option 4

- In multiplying the frequency of the transient with the unavailability of the safety system we have in fact abolished the single failure criteria which require us to consider the most limiting failure
- Therefore we need to somehow compensate for it by hardware solutions or by using tools and principles from RIDM process

Option 4

- In addition, in order to preserve DiD, we need to repeat the calculation for the next most unfavourable single failure and demonstrate the fulfilment of the original acceptance criteria
- How to find the next most unfavourable single failure ?

Fig. 8 Hypothetical Event Tree



Frequency	Category of the Sequence	Importance in terms of available safety margins
F1	2	
F2	2	III
F3	3	IV
F4	4	
F5	3	I
F6	4	
F7	4	
F8	5	
F9	3	II
F10	3	
F11	4	
F12	4	
F13	4	
F14	5	
F15	4	
F16	5	

Option 4

- From such tree we can extract sequences with single failures f2, f3, f5 and f9
- By performing Option 1,2 or 3 calculations for these sequences we calculate safety margins
- In this way we can rank sequences and the sequence with the least safety margin is the most unfavourable one

Option 4

- In DSA all these sequences must fulfil the acceptance criteria corresponding to the category of the Initiating Event (Cat. 2)
- In Fig 8 they however fall into Cat. 2 or 3 depending on their frequency
- Let us suppose that by using Option 1,2 or 3 calculations we have ranked them as f5, f9, f2 and f3 with regard to available safety margins

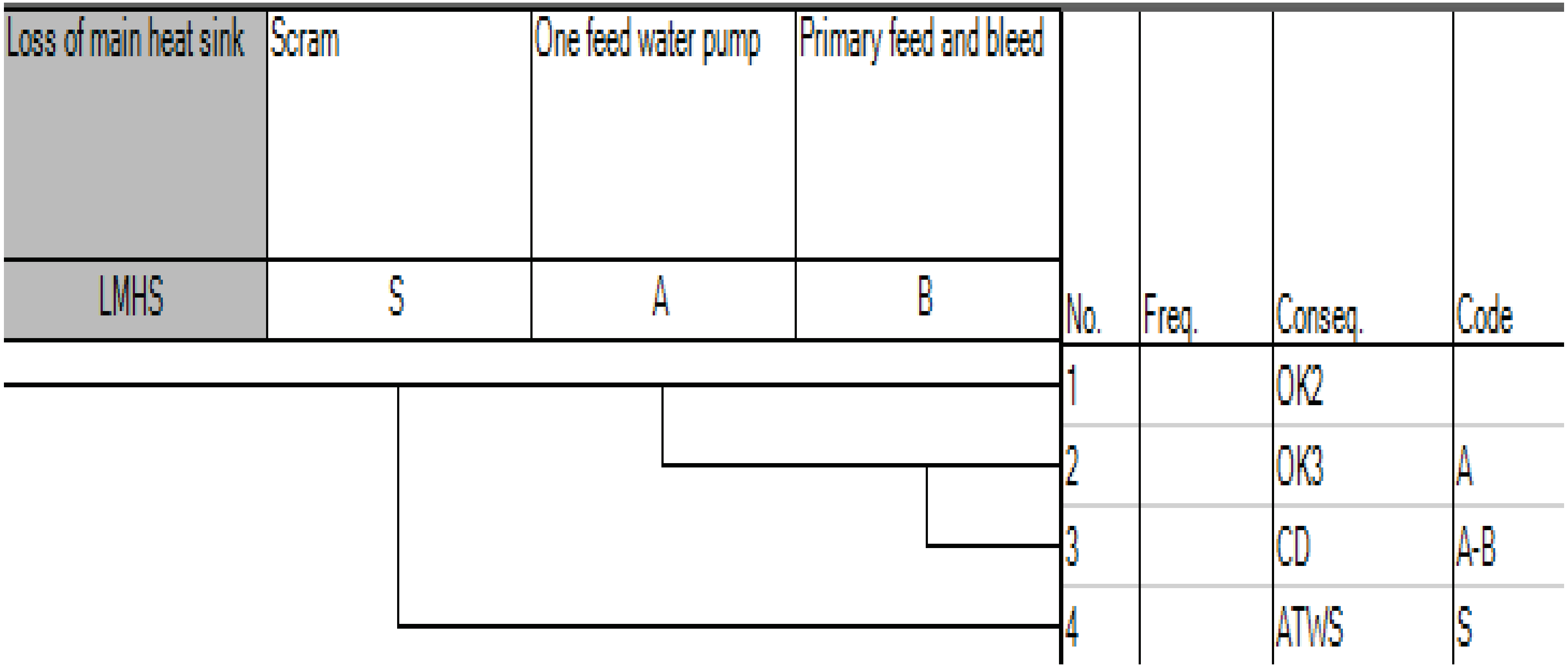
Option 4

- Using Option 4 we relax the single failure criteria for f5 and verify that it satisfies the acceptance criteria for Cat. 3 and apply “compensatory measures” - for violating single failure criterion
- Now we need to verify that f2, f3 and f9 satisfy the acceptance criteria for Cat. 2
- In case one of them would fail to do so, we can apply Option 4 as in the case of f5 but in any case f2 must satisfy acceptance criteria for Cat. 2

Example of Compensatory Measures

- **Example: Loss of Main Heat Sink (following the Turbine Trip) after the Power Uprate**
- **To handle this event the shutdown system (S) and one of two feed water pumps (A) is required**
- **IE LMHS puts the plant in state OK2 – abnormal operation. If S and A available, plant stays in OK2 (sequence 1 in Fig 9)**
- **If both pumps fail, primary feed and bleed required and that puts the plant in state OK3 (sequence 2 in Fig 9)**

Fig 9: ET for LMHS before PU

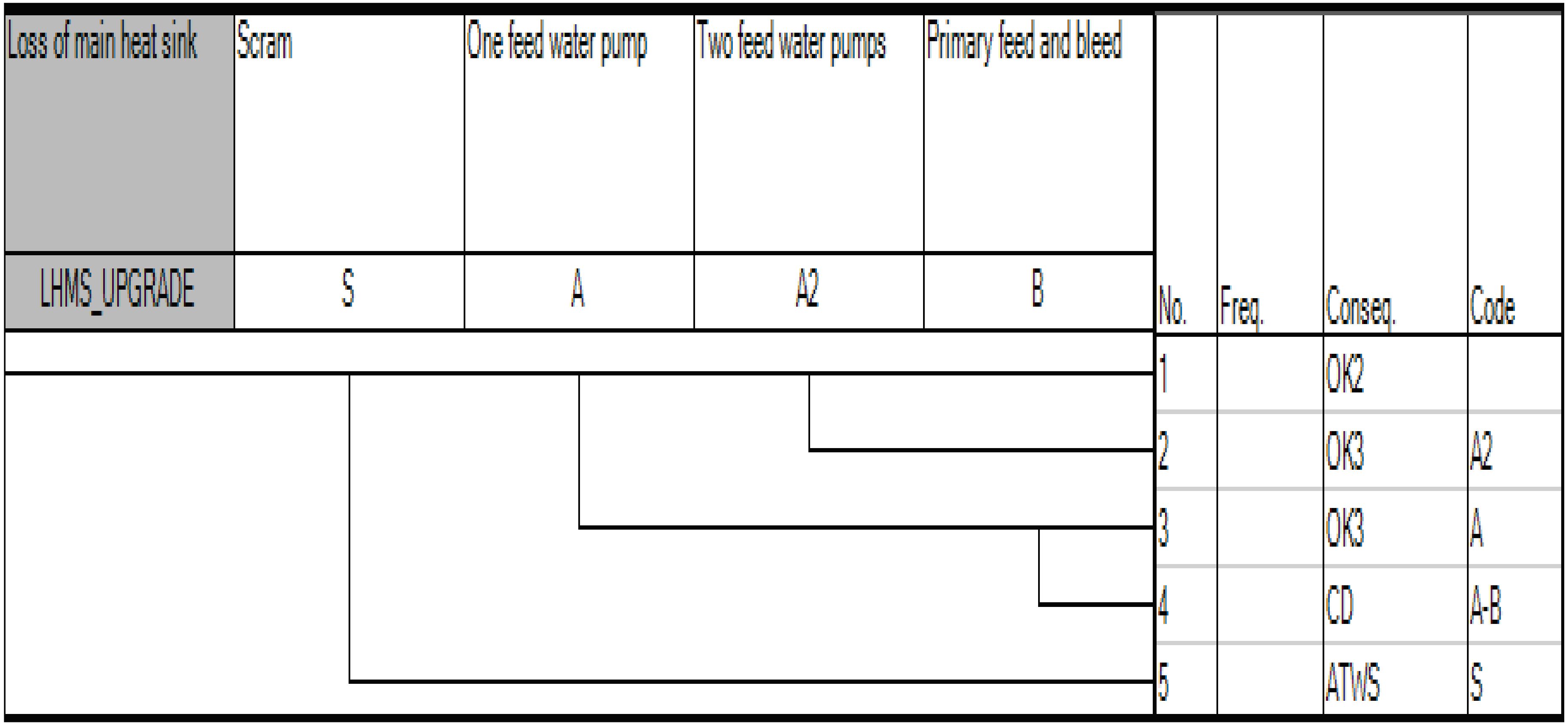


No.	Freq.	Conseq.	Code
1		OK2	
2		OK3	A
3		CD	A-B
4		ATWS	S

Example – cont.

- After the PU residual heat is too large to be removed using only one pump
- In case only one pump available pzs. safety valve has to open and close several times – plant state OK3 - (sequence 2 in Fig. 10)
- So the plant is no longer single failure proven
- We apply Option 4; multiply frequency of IE with the probability to have two pumps unavailable (sequence 2 in Fig.10)
- This brings us from OK2 to a new plant condition OK3 – releases in OK2 are the same as for normal operation and larger in OK3 due to opening of prz. safety valve and venting of containment
- Multiplied frequency is sufficiently low so that OK3 is acceptable in spite of violating the single failure criteria

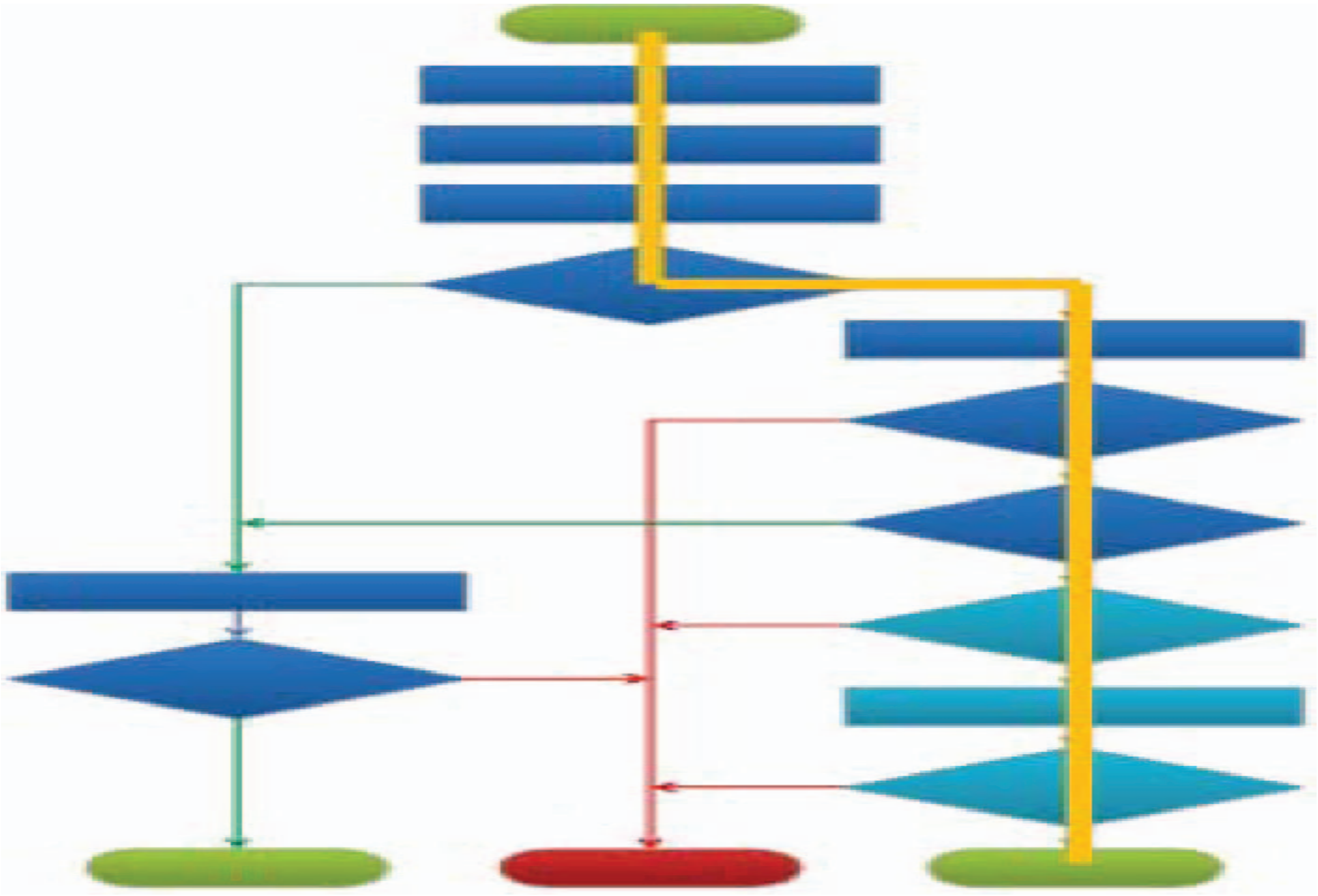
Fig. 10: ET for LMHS after the PU



Loss of main heat sink	Scram	One feed water pump	Two feed water pumps	Primary feed and bleed
LHMS_UPGRADE	S	A	A2	B

No.	Freq.	Conseq.	Code
1		OK2	
2		OK3	A2
3		OK3	A
4		CD	A-B
5		ATWS	S

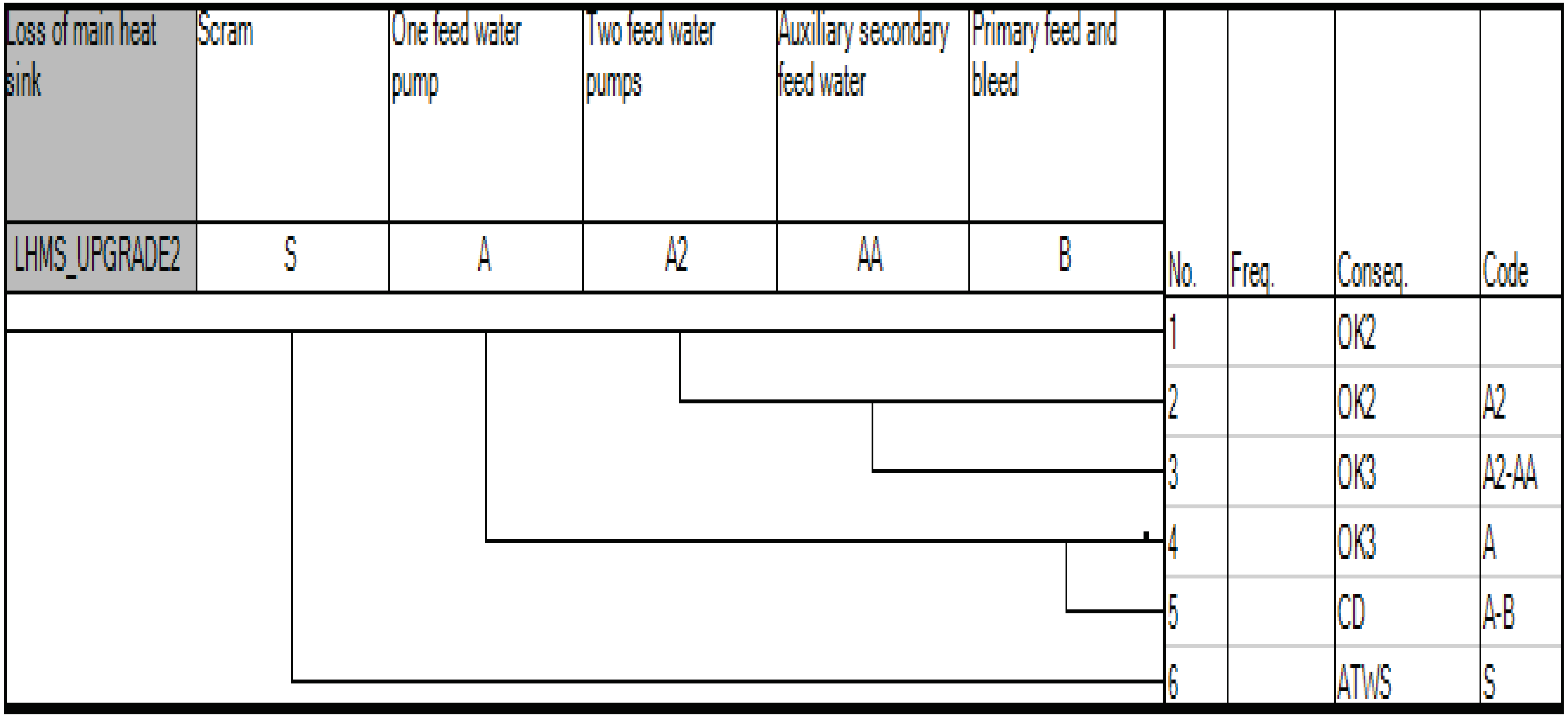
Figure 11: Applying Option 4 sequence 2 can be reclassified, if its frequency is low enough. Then it meets the acceptance limits of the next higher class.



Example - cont.

- To compensate for not reaching OK2 and violating the single failure criteria an auxiliary feed water system might be added
- This system is not single failure proven and is on its own not capable to remove the residual heat
- Introduction of this new auxiliary system “brings back” the single failure robustness for System A – with only one feed water pump plus the new auxiliary system we stay in OK2 (sequence 2 in Fig 12)

Fig 12: ET for LMHS after PU with add. AFWS



Loss of main heat sink LHMS_UPGRADE2	Scram S	One feed water pump A	Two feed water pumps A2	Auxiliary secondary feed water AA	Primary feed and bleed B	No.	Freq.	Conseq.	Code
						1		OK2	
						2		OK2	A2
						3		OK3	A2-AA
						4		OK3	A
						5		CD	A-B
						6		ATWS	S

Option 4 and LOCA redefinition

- **Risk-Informed changes to LOCA Tech. Requirements (10CFR50.46a, Dec,2010)**
- **Introduction of Transition Break Size (TBS)**
- **LOCA < TBS Cat 4a DBA**
- **LOCA > TBS Cat 4b BDBA**

Figure 13: Current procedure for LOCA mapped to Option 4 (left for LOCA above TBS, right for LOCA below TBS). For LOCA below TBS, Preclassify is effectively 0.

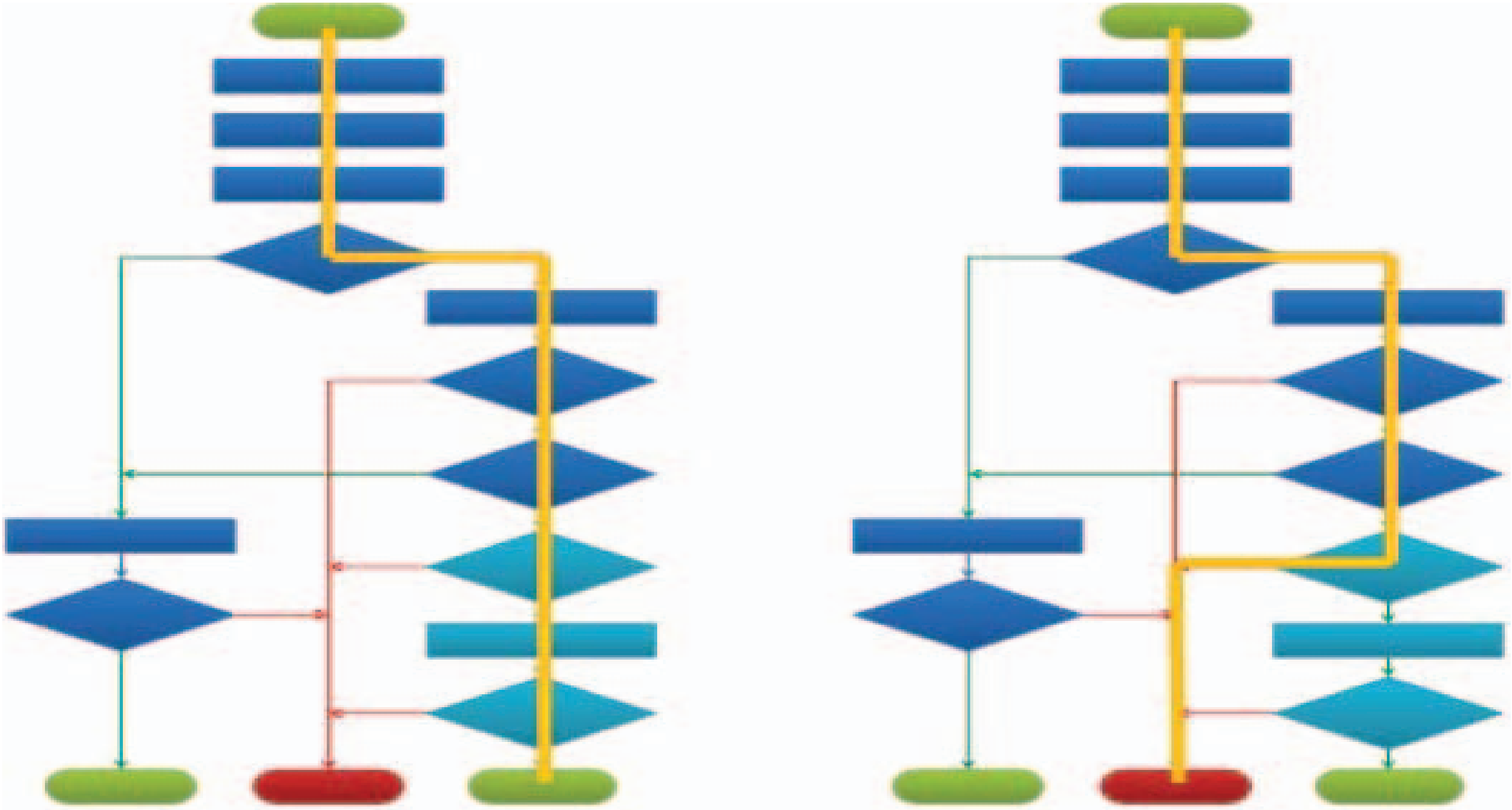
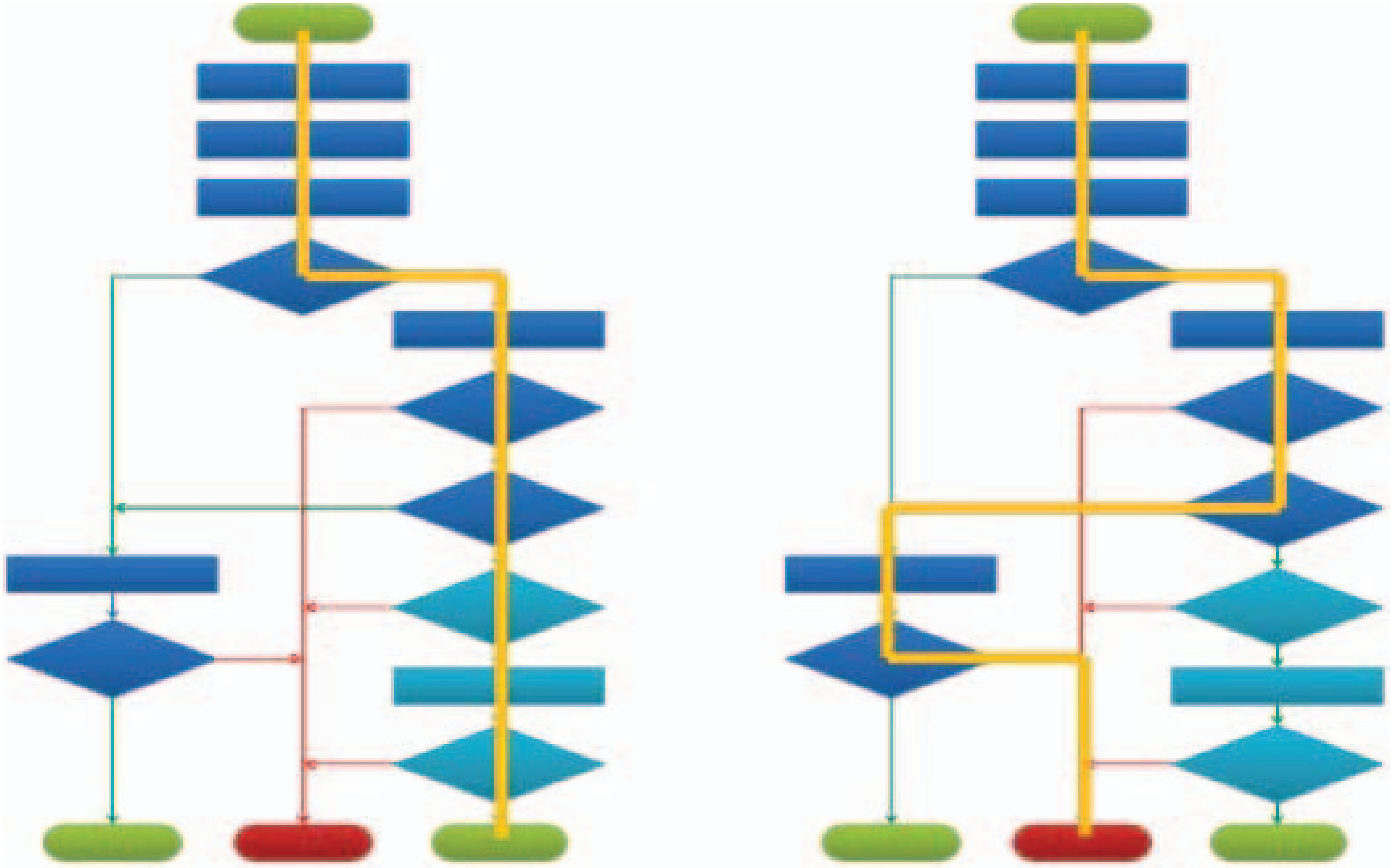


Figure 14: Option 4 applied to LBLOCA. Sequences with single failure can be reclassified if Pseq < Preclassify (left). All sequences fulfilling the acceptance criterion for their class have to also fulfil the tightened criterion for the next higher class otherwise the design has to be changed (right).



Combined Use of PSA and BEPU in Design Extension

- WENRA Reactor Safety Reference Levels. Jan. 2008
- Design extension = measures taken to cope with additional events or combination of events, not foreseen in the design of the plant
- Such measures need not to involve conservative engineering practices

Combined Use of PSA and BEPU in Design Extension

Flooding scenario:

- As water level increases, the cutsets of the systems needed to attain a safe shutdown will eventually turn failed
- Cutsets + EOPs + SAMGs will determine the dynamic of the accident evolution – therefore both PSA and BEPU are needed
- The BDBA damage categories are predefined (Cat 1 – core damage, Cat 2 – core damage + early release, Cat 3 – Core damage + late releases
- Acceptance criteria for each category need to be determined, based on probabilities of escalation from one to the next category

Conclusions

The main characteristics of Option 4:

- Availability of safety systems is determined using probabilistic arguments
- Acceptance limit relates to the frequency of the sequence following each PIE
- Requirements for sequence runs that do not fulfill the 95/95 acceptance criterion are tightened. In doing so any cliff-edge effects are excluded
- Certain sequences having low enough probability of occurrence can be reclassified into the next higher class, where they would be compared to the acceptance limit for that class (or some tightened limit) – relaxing a single failure criterion

Conclusions – cont.

General Conclusions:

- Conservative approach is still used for licensing purposes.
- Some countries have moved to BEPU approach.
- Option 4 will remain a “research” option for a quite some time.
- Nevertheless the combined insights from DAS and PSA will continue to be explored as recommended also in the IAEA Requirements on Safety Assessment for Facilities and Activities (GS-R-4)

Extension of BEPU Methods to Sub-channel Thermal-Hydraulics and to Coupled Three-Dimensional Neutronics/Thermal-Hydraulics Codes

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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*
- *Uncertainty Quantification of Sub-Channel Thermal-Hydraulic Calculations*
- *Propagation of Uncertainties in Coupled Three-Dimensional Neutronics/Thermal-Hydraulic Calculations*
- *Conclusions*

Introduction

- As compared to the traditional conservative calculations the BEPU evaluations result in increased plant operation flexibility and improved performance:
  - Knowledge of uncertainties in input parameters and data;
  - Knowledge and understanding of sources and uncertainties and biases in analytic and numerical modelling approximations;
  - Comprehensive methodologies to propagate these uncertainties through calculations to predictions of interest.
- Historically two main approaches have been adopted in the nuclear engineering applications:
  - Statistical methods, which are utilizing statistical sampling based on statistical propagation and processing. The statistical methods have been applied to thermal-hydraulics calculations;
  - Deterministic methods, which are using sensitivity analysis based on first-order perturbation theory. Deterministic methods have been used for multi-group cross-section uncertainty propagation in multiplication factor (k_{eff}) predictions for criticality neutronics calculations.

Introduction

- In statistical methods in order to incorporate uncertainties into the process usually many runs of the computer code are required and the result is a range of results with associated probabilities:
  - The best choice has been indentified in the use of Order Statistics;
  - The extension of BEPU methods to the sub-channel thermal-hydraulic codes is illustrated in this presentation on the example of the Pennsylvania State University (PSU) version of COBRA-TF (CTF) in cooperation with GRS using the SUSA package.
- The extension of BEPU to coupled 3-D neutronics/thermal-hydraulic calculations requires the following capabilities:
  - Higher than linear order uncertainty analysis techniques capable of treating the non-linear thermal-hydraulic feedback phenomena;
  - Approaches allowing for combination of different input sources of uncertainties;
  - Computationally efficient methods in dealing with large size of input parameters often associated with realistic reactor models.

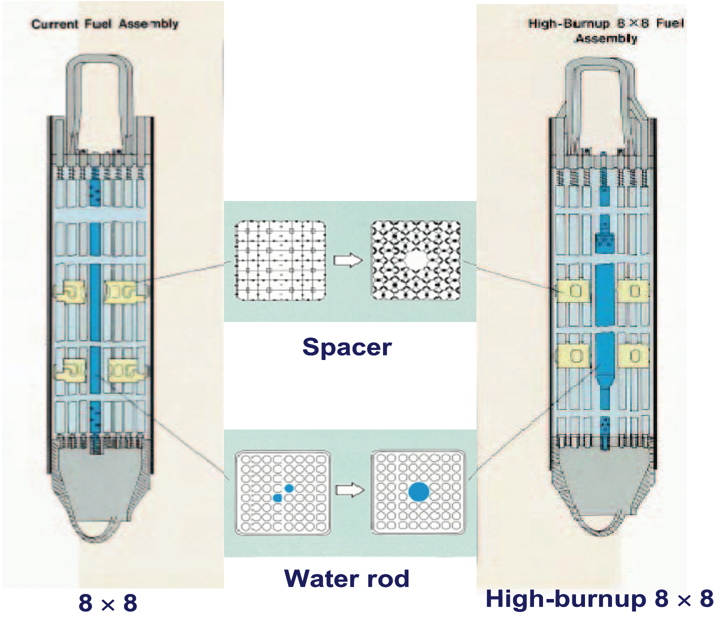
Extension of BEPU to Sub-channel Codes

The sub-channel code CTF (COBRA-TF) features three fields representation of two-phase flow → set of nine time-averaged conservation equations written in a semi-implicit form using a donor cell differencing for the convective quantities

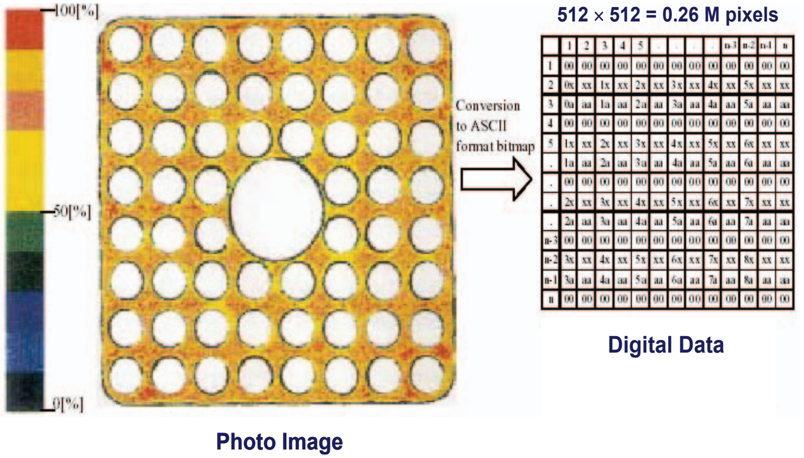
The CTF applications to the OECD/NRC BFBT Benchmark have indicated an over-prediction of the bundle void fraction, which coincided with a slightly over-predicted total two-phase pressure drop

The cause of both phenomena was believed to be overestimated interfacial drag forces leading to an overestimation of the slip and subsequently under-predicted vapor velocity yielding a higher void fraction

Uncertainty and sensitivity analysis were performed in the framework of the OECD/NRC BFBT benchmark



BFBT assembly types



Data resolution

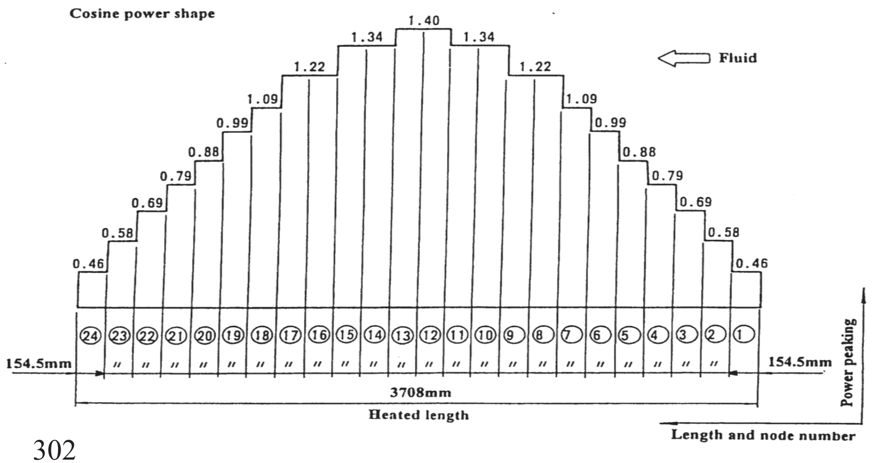
Analyzed BFBT Test Cases

The selected cases for Exercise I-4 in Volume II of the BFBT Specification are given below

The output parameter is the subchannel void fraction

Test Case	*Void Fraction (%)*	*Pressure (MPa)*	*Flow Rate (t/h)*	*Inlet Subcooling (kJ/kg)*	*Inlet Temperature (ºC)*	*Power (MW)*
Test 4101-02	57.1	0.994	10.12	53.3	167.5	0.32
Test 4101-13	86.8	1.224	55.01	92.5	167.9	4.46
Test 4101-69	18.2	8.638	10.08	52.5	291.1	0.23
Test 4101-86	69.8	8.705	54.59	54.2	291.3	4.62

1.15	1.30	1.15	1.30	1.30	1.15	1.30	1.15
1.30	0.45	0.89	0.89	0.89	0.45	1.15	1.30
1.15	0.89	0.89	0.89	0.89	0.89	0.45	1.15
1.30	0.89	0.89			0.89	0.89	1.15
1.30	0.89	0.89			0.89	0.89	1.15
1.15	0.45	0.89	0.89	0.89	0.89	0.45	1.15
1.30	1.15	0.45	0.89	0.89	0.45	1.15	1.30
1.15	1.30	1.15	1.15	1.15	1.15	1.30	1.15

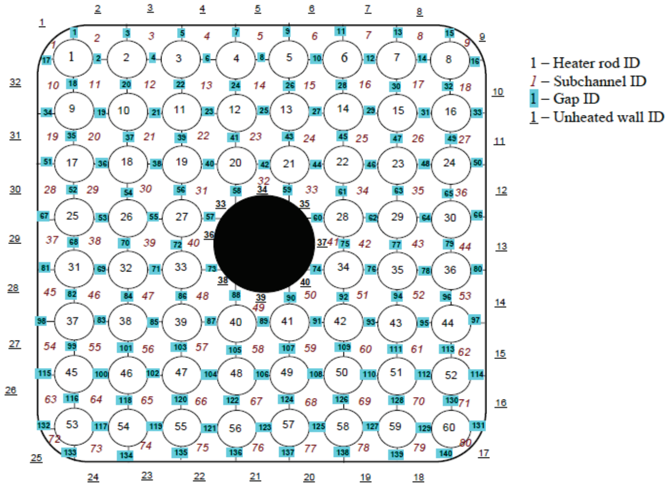


Extension of BEPU to Sub-channel Codes

- **The GRS methodology has been already applied to a wide field of different applications – in this study it is applied to BWR steady-state subchannel void predcitions**
- **Two-sided tolerance intervals are used and the sample size N is selected to be 153 for β=0.95 and γ=0.95**
- **Twelve (12) uncertain input parameters have been selected - the selection was guided by the Phenomena Identification and Ranking Tables (PIRT):**
  - ✓ ***F. Aydogan, L. Hochreiter, K. Ivanov, M. Martin, H. Utsuno, E. Sartori, 2007. NUPEC BWR Full Size Fine-Mesh Bundle Test (BFBT) benchmark, Volume II: Uncertainty and sensitivity analyses of void distribution and critical power-prediction. NEA/NSC/DOC(2007)21.***
- **The developed calculation model with CTF for high-burnup 8×8 BWR assembly consists of 80 sub-channels, 40 axial nodes, and 140 transverse connections between sub-channels (gaps).**

Uncertain Input Parameters

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



CTF full assembly model nodalization - 80 sub-channels each divided into 40 axial nodes

Accuracy Analysis

For a series of N individual assessments by a code (x_{code}^{n}) of a output variable x, the following statistical comparison with experimental data were performed:

Mean value:

$$x^{code} = \frac{1}{N} \sum_{n \le N} x_n^{code}$$

Individual relative bias error:

$$\delta_n^{code} = \frac{x_n^{code} - x^{\exp}}{x^{\exp}}$$

Mean relative bias error:

$$\delta^{code} = \frac{1}{N} \sum_{n \le N} \delta_n^{code}$$

Maximum bias error:

$$\varepsilon^{code} = \max \left| \delta_n^{code} \right|$$

Absolute error:

$$\Delta_n^{code} = x_n^{code} - x^{\exp}$$

Coverage ratio:

$$R^{\text{code}} = \frac{card\left\{ \Delta_n^{\text{code}} \quad \text{with} \quad \Delta_n^{\text{code}} \le \varepsilon^{\exp} \right\}}{N}$$

Standard deviation:

$$\sigma^{code} = \pm \sqrt{\frac{1}{N} \sum_{n \le N} (x_n^{code} - x^{code})^2}$$

Accuracy Analysis

Case 4101-13

Calculated void distribution

0.88	0.89	0.90	0.89	0.89	0.89	0.90	0.89	0.87
0.89	0.90	0.89	0.90	0.90	0.89	0.90	0.91	0.89
0.90	0.89	0.88	0.88	0.87	0.87	0.88	0.89	0.89
0.89	0.90	0.88	0.85	0.96	0.85	0.87	0.88	0.88
0.90	0.90	0.87	0.95		0.96	0.87	0.89	0.88
0.89	0.89	0.86	0.87	0.94	0.86	0.86	0.88	0.88
0.90	0.90	0.88	0.86	0.87	0.86	0.88	0.89	0.89
0.89	0.91	0.89	0.89	0.89	0.88	0.89	0.91	0.88
0.87	0.89	0.89	0.88	0.88	0.88	0.89	0.88	0.87

Measured void distribution

0.81	0.82	0.83	0.87	0.88	0.86	0.87	0.87	0.82
0.85	0.87	0.89	0.90	0.89	0.92	0.91	0.91	0.84
0.85	0.91	0.89	0.86	0.88	0.86	0.85	0.88	0.85
0.86	0.90	0.86	0.89	0.82	0.87	0.85	0.89	0.85
0.88	0.91	0.85	0.78		0.76	0.86	0.91	0.89
0.88	0.89	0.87	0.87	0.78	0.85	0.84	0.87	0.88
0.87	0.89	0.87	0.83	0.88	0.85	0.85	0.88	0.90
0.85	0.87	0.89	0.86	0.88	0.88	0.89	0.88	0.88
0.75	0.86	0.88	0.88	0.87	0.87	0.86	0.84	0.79

Absolute Error

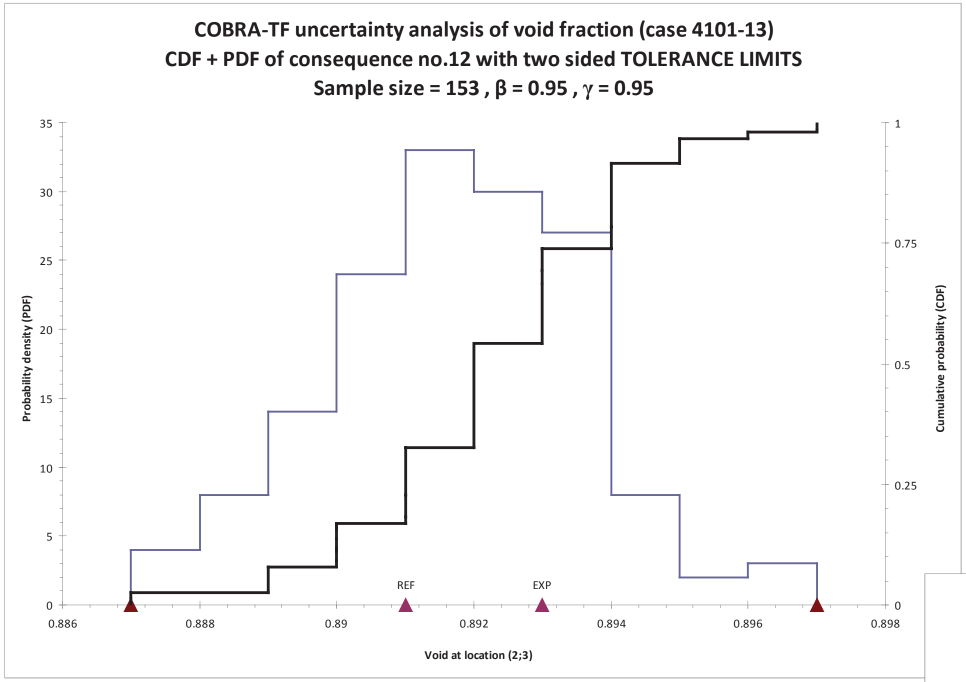
0.07	0.07	0.07	0.02	0.01	0.03	0.03	0.02	0.05
0.04	0.03	0.00	0.00	0.01	-0.03	-0.01	0.00	0.05
0.05	-0.02	-0.01	0.02	-0.01	0.01	0.03	0.01	0.04
0.03	0.00	0.02	-0.04	0.14	-0.02	0.02	-0.01	0.03
0.02	-0.01	0.02	0.17		0.20	0.01	-0.02	-0.01
0.01	0.00	-0.01	0.00	0.16	0.01	0.02	0.01	0.00
0.03	0.01	0.01	0.03	-0.02	0.01	0.03	0.01	-0.01
0.04	0.04	0.00	0.03	0.01	0.00	0.00	0.03	0.00
0.12	0.03	0.01	0.00	0.01	0.01	0.03	0.04	0.08

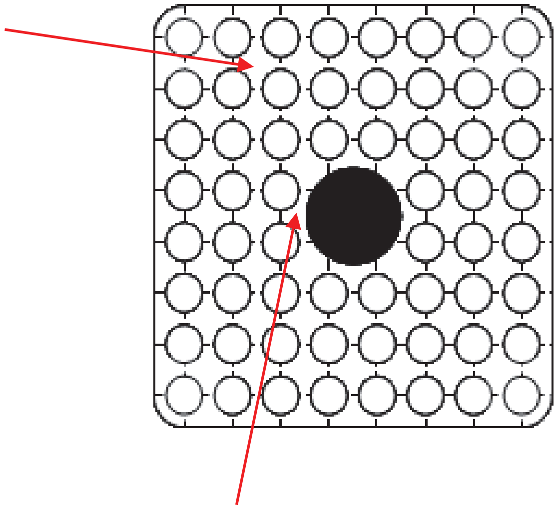
Coverage Ratio with ε^{exp} = 0.08

1.00	0.89	1.00	1.00	1.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	0.01	1.00	1.00	1.00	1.00
1.00	1.00	1.00	0.00		0.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	0.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
0.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	0.67

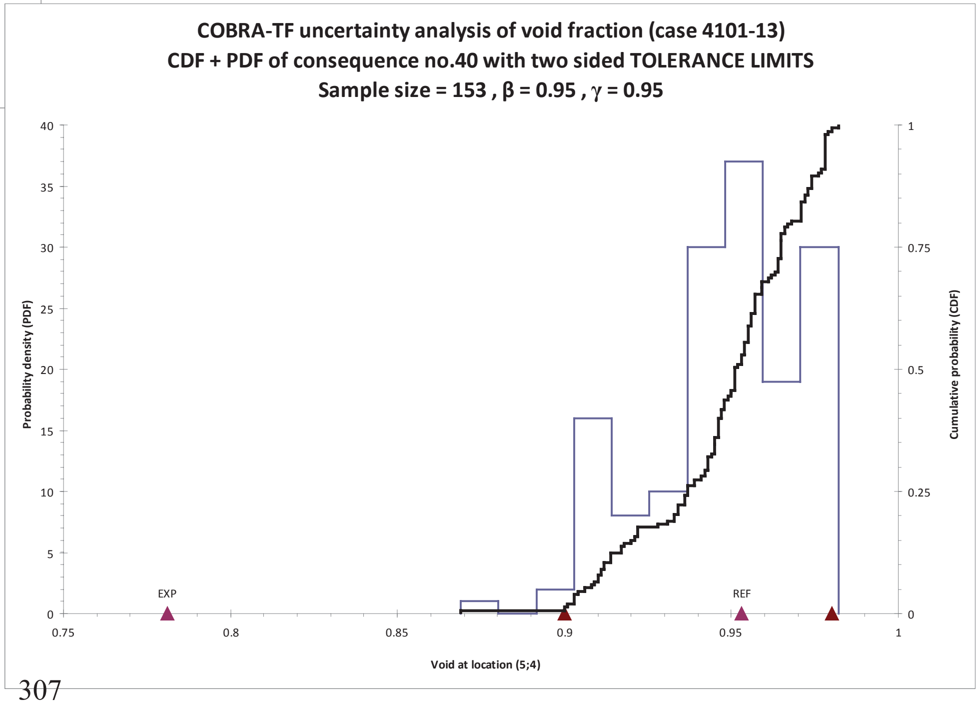
Uncertainty Analysis

Case 4101-13


COBRA-TF uncertainty analysis of void fraction (case 4101-13)
CDF + PDF of consequence no.12 with two sided TOLERANCE LIMITS
Sample size = 153 , β = 0.95 , γ = 0.95



Void Fraction (%)	*Pressure (MPa)*	*Flow Rate (t/h)*	*Power (MW)*
86.8	1.224	55.01	4.46


COBRA-TF uncertainty analysis of void fraction (case 4101-13)
CDF + PDF of consequence no.40 with two sided TOLERANCE LIMITS
Sample size = 153 , β = 0.95 , γ = 0.95

Accuracy Analysis

Case 4101-69

Calculated void distribution

0.15	0.21	0.20	0.20	0.20	0.18	0.16	0.13	0.15
0.21	0.22	0.21	0.22	0.22	0.21	0.20	0.21	0.13
0.20	0.21	0.19	0.20	0.20	0.19	0.19	0.21	0.16
0.20	0.22	0.20	0.18	0.16	0.18	0.18	0.20	0.17
0.20	0.22	0.20	0.16		0.18	0.20	0.21	0.18
0.18	0.21	0.18	0.18	0.17	0.17	0.19	0.21	0.19
0.16	0.20	0.19	0.18	0.20	0.19	0.18	0.21	0.20
0.13	0.21	0.21	0.20	0.21	0.21	0.21	0.24	0.21
0.15	0.13	0.16	0.17	0.18	0.19	0.20	0.21	0.15

Measured void distribution

0.09	0.14	0.12	0.14	0.14	0.14	0.15	0.16	0.13
0.16	0.25	0.19	0.22	0.24	0.21	0.23	0.27	0.18
0.14	0.23	0.22	0.21	0.21	0.18	0.17	0.22	0.15
0.14	0.22	0.20	0.15	0.10	0.15	0.20	0.21	0.13
0.12	0.23	0.22	0.09		0.09	0.18	0.22	0.13
0.14	0.21	0.20	0.15	0.07	0.16	0.18	0.21	0.13
0.15	0.21	0.20	0.20	0.17	0.19	0.20	0.22	0.17
0.15	0.21	0.24	0.22	0.21	0.21	0.20	0.24	0.17
0.10	0.18	0.16	0.13	0.14	0.13	0.13	0.16	0.14

Absolute Error

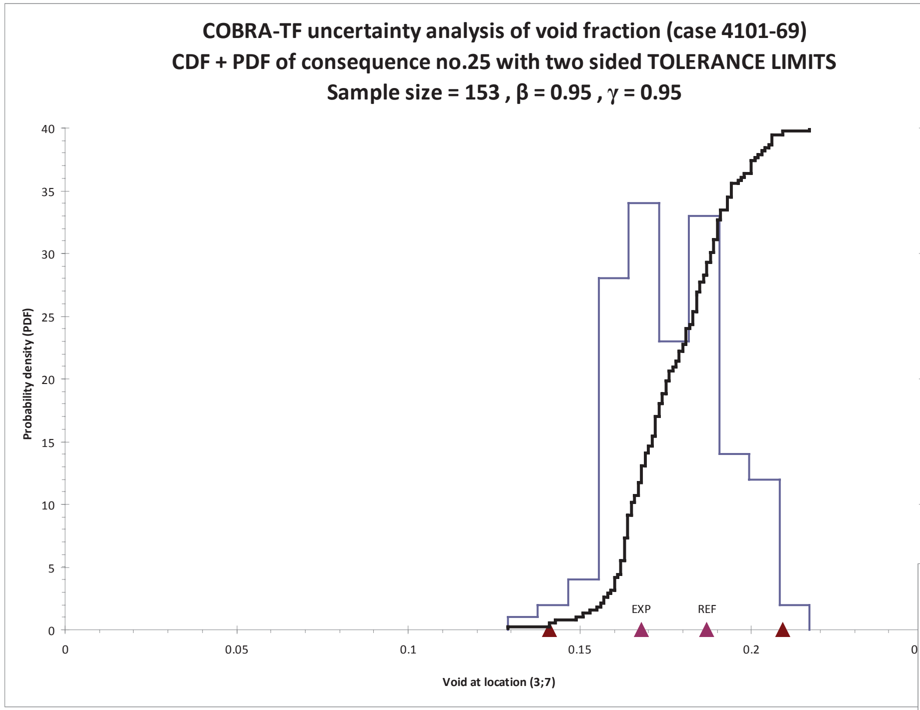
0.06	0.07	0.08	0.07	0.06	0.04	0.00	-0.03	0.02
0.06	-0.03	0.02	0.00	-0.02	-0.01	-0.03	-0.06	-0.04
0.06	-0.02	-0.04	-0.02	-0.01	0.01	0.02	-0.02	0.02
0.06	-0.01	0.00	0.03	0.06	0.03	-0.02	-0.01	0.04
0.08	-0.01	-0.03	0.07		0.09	0.02	-0.01	0.05
0.04	0.00	-0.01	0.03	0.09	0.01	0.02	0.00	0.06
0.01	0.00	-0.01	-0.01	0.03	0.00	-0.01	-0.01	0.03
-0.02	0.00	-0.03	-0.02	0.00	-0.01	0.01	0.00	0.04
0.05	-0.05	0.01	0.04	0.04	0.06	0.07	0.05	0.00

Coverage Ratio with $\varepsilon^{exp} = 0.08$

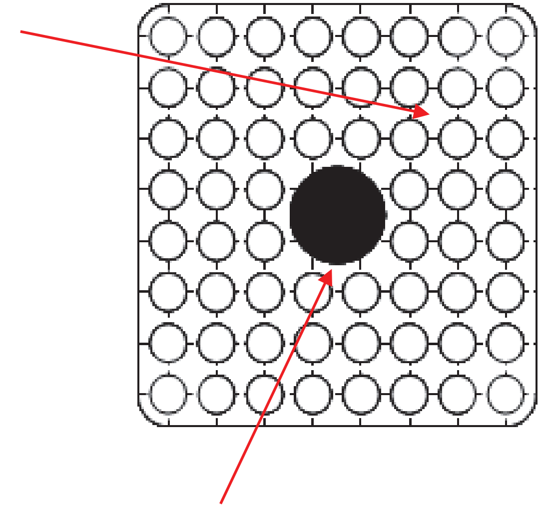
0.67	0.76	0.71	0.82	0.90	0.99	0.97	0.87	0.83
0.80	0.88	0.97	0.95	0.93	0.99	0.97	0.96	0.95
0.77	0.89	0.95	0.99	0.99	0.98	1.00	0.98	0.97
0.88	0.92	1.00	0.99	0.90	0.91	0.86	0.99	0.99
0.69	0.97	0.89	0.76		0.59	0.95	1.00	0.99
0.97	1.00	0.88	0.86	0.25	0.83	0.92	0.99	0.93
0.99	1.00	1.00	0.99	1.00	0.88	1.00	1.00	1.00
0.89	0.99	0.99	1.00	1.00	0.99	1.00	1.00	1.00
0.84	0.99	0.99	0.93	0.97	0.88	0.76	0.99	0.82

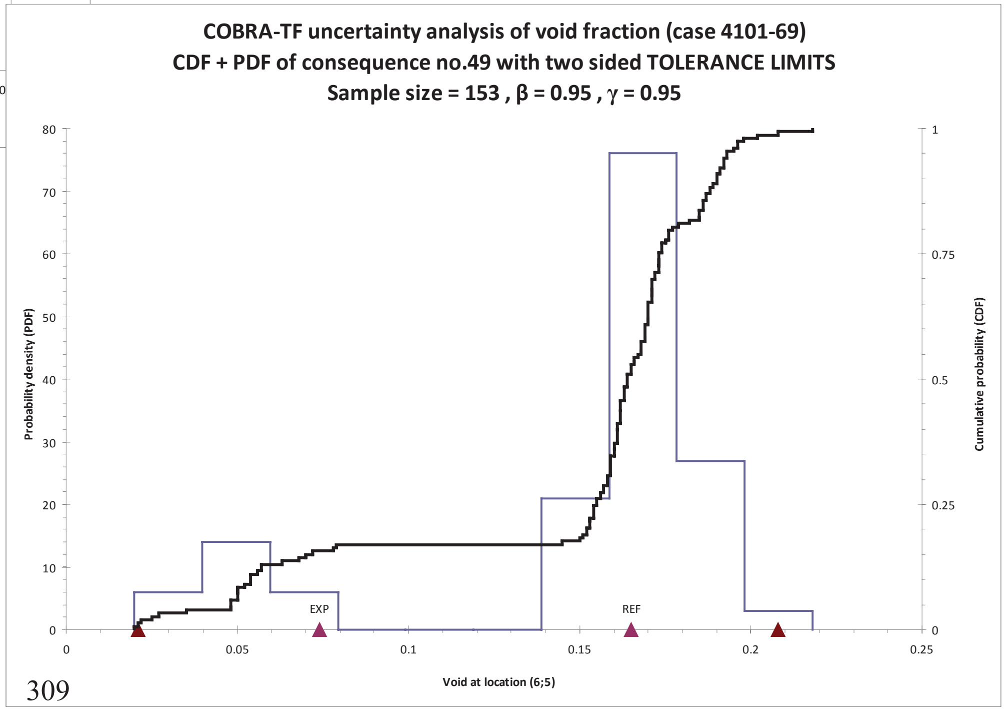
Uncertainty Analysis

Case 4101-69



COBRA-TF uncertainty analysis of void fraction (case 4101-69)
CDF + PDF of consequence no.25 with two sided TOLERANCE LIMITS
Sample size = 153 , β = 0.95 , γ = 0.95





COBRA-TF uncertainty analysis of void fraction (case 4101-69)
CDF + PDF of consequence no.49 with two sided TOLERANCE LIMITS
Sample size = 153 , β = 0.95 , γ = 0.95

Void Fraction (%)	Pressure (MPa)	Flow Rate (t/h)	Power (MW)
18.2	8.638	10.08	0.23

Accuracy Analysis

Case 4101-86

Calculated void distribution

0.68	0.72	0.71	0.72	0.72	0.72	0.71	0.72	0.68
0.72	0.78	0.77	0.77	0.77	0.77	0.77	0.77	0.72
0.71	0.77	0.77	0.77	0.76	0.76	0.77	0.77	0.71
0.72	0.77	0.77	0.75	0.69	0.75	0.76	0.77	0.71
0.72	0.77	0.76	0.69		0.69	0.76	0.76	0.71
0.72	0.77	0.77	0.73	0.68	0.73	0.76	0.77	0.71
0.71	0.77	0.77	0.77	0.76	0.76	0.77	0.77	0.71
0.72	0.78	0.77	0.77	0.77	0.77	0.77	0.77	0.72
0.68	0.72	0.71	0.72	0.71	0.72	0.71	0.72	0.68

Measured void distribution

0.66	0.72	0.68	0.73	0.75	0.73	0.70	0.76	0.75
0.76	0.74	0.73	0.74	0.72	0.71	0.69	0.76	0.76
0.74	0.73	0.68	0.65	0.66	0.67	0.66	0.75	0.71
0.74	0.71	0.67	0.68	0.55	0.63	0.63	0.66	0.69
0.74	0.74	0.66	0.56		0.52	0.68	0.72	0.71
0.75	0.73	0.70	0.68	0.53	0.65	0.67	0.66	0.69
0.74	0.73	0.69	0.61	0.63	0.65	0.68	0.70	0.75
0.77	0.73	0.69	0.71	0.70	0.69	0.70	0.72	0.79
0.67	0.77	0.73	0.72	0.69	0.66	0.68	0.77	0.72

Absolute Error

0.02	0.00	0.03	-0.01	-0.03	-0.01	0.01	-0.04	-0.07
-0.04	0.04	0.04	0.03	0.05	0.06	0.08	0.01	-0.05
-0.03	0.04	0.09	0.12	0.10	0.10	0.10	0.02	0.00
-0.02	0.06	0.10	0.07	0.13	0.12	0.13	0.10	0.02
-0.02	0.03	0.10	0.13		0.17	0.08	0.05	0.00
-0.03	0.04	0.07	0.05	0.16	0.08	0.09	0.11	0.02
-0.02	0.04	0.08	0.16	0.13	0.11	0.09	0.08	-0.04
-0.05	0.05	0.08	0.06	0.06	0.07	0.07	0.05	-0.07
0.01	-0.05	-0.02	0.00	0.02	0.05	0.03	-0.05	-0.04

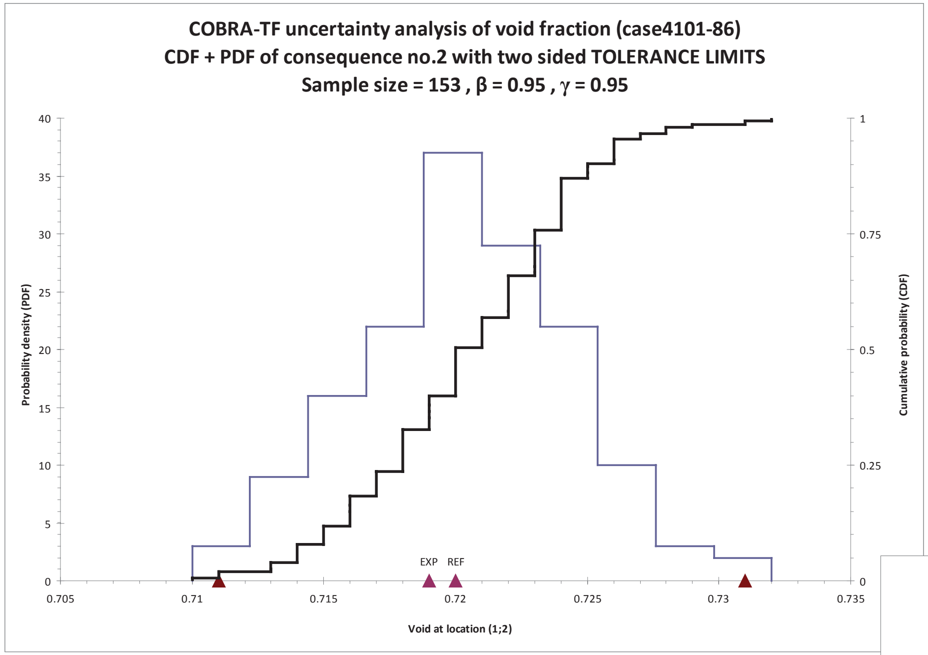
Coverage Ratio with $\varepsilon^{exp} = 0.08$

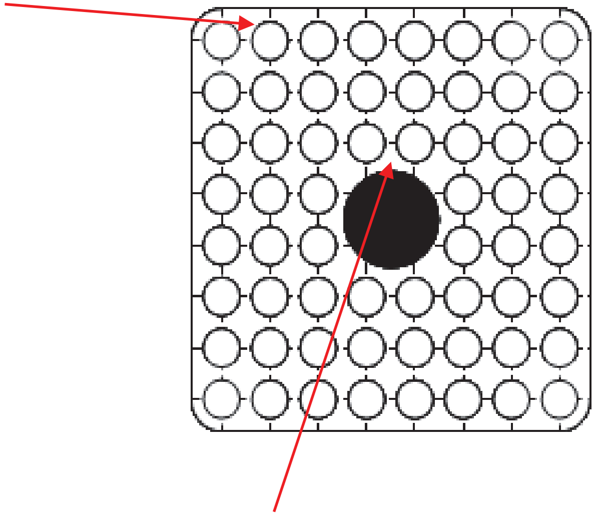
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	0.99
1.00	1.00	1.00	1.00	1.00	1.00	0.22	1.00	1.00
1.00	1.00	0.00	0.00	0.00	0.00	0.00	1.00	1.00
1.00	1.00	0.00	0.92	0.00	0.00	0.00	0.00	1.00
1.00	1.00	0.00	0.00		0.00	0.32	1.00	1.00
1.00	1.00	1.00	1.00	0.00	0.63	0.01	0.00	1.00
1.00	1.00	0.82	0.00	0.00	0.00	0.00	0.92	1.00
1.00	1.00	0.49	1.00	1.00	0.99	1.00	1.00	0.98
1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

Uncertainty Analysis

Case 4101-86

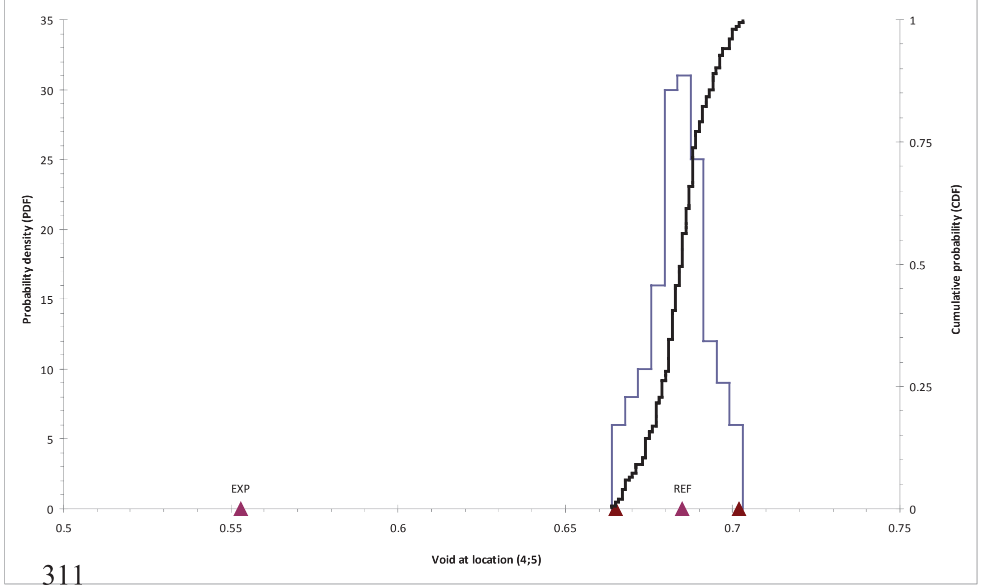
COBRA-TF uncertainty analysis of void fraction (case4101-86)
CDF + PDF of consequence no.2 with two sided TOLERANCE LIMITS
Sample size = 153 , β = 0.95 , γ = 0.95





Void Fraction (%)	Pressure (MPa)	Flow Rate (t/h)	Power (MW)
69.8	8.705	54.59	4.62

COBRA-TF uncertainty analysis of void fraction (case4101-86)
CDF + PDF of consequence no.32 with two sided TOLERANCE LIMITS
Sample size = 153 , β = 0.95 , γ = 0.95



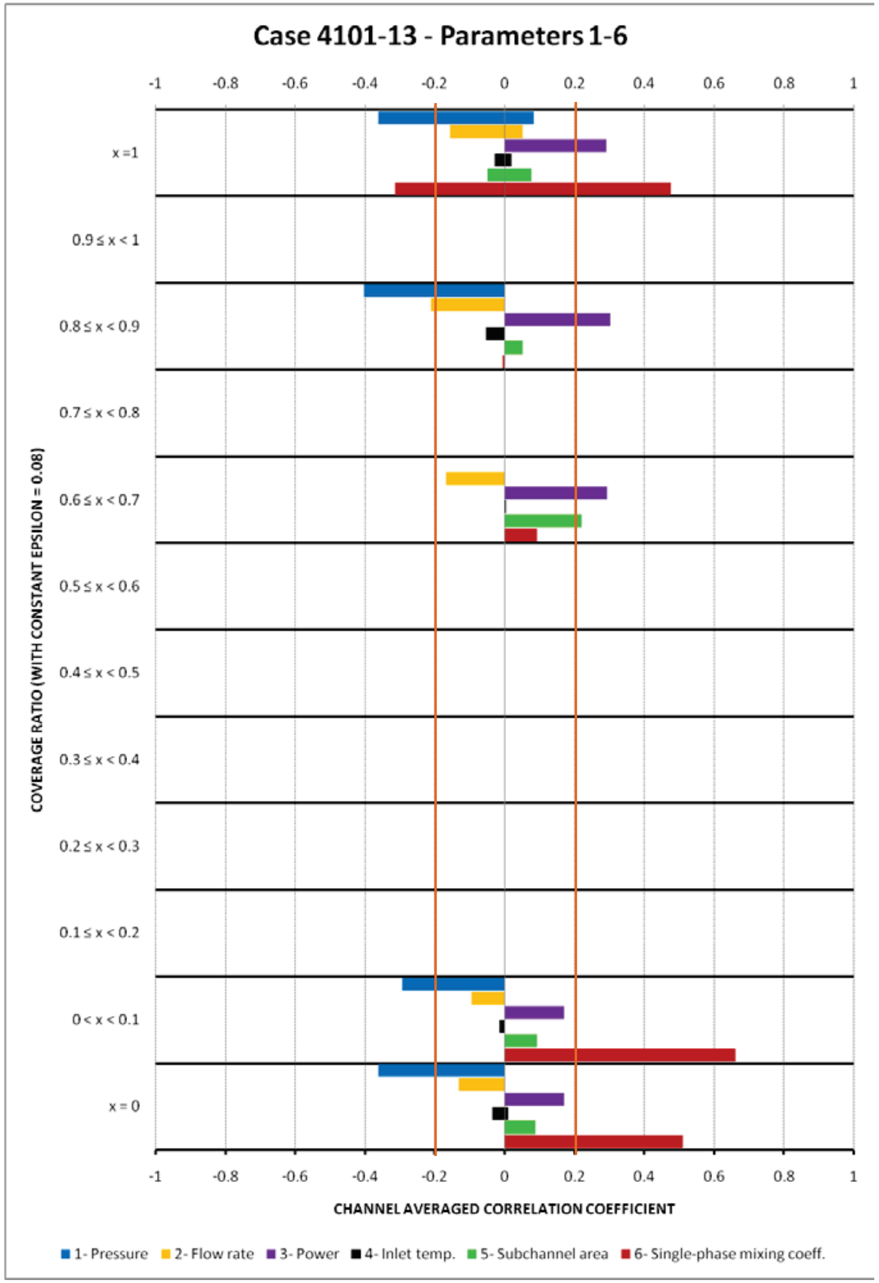
Accuracy/Uncertainty Analysis

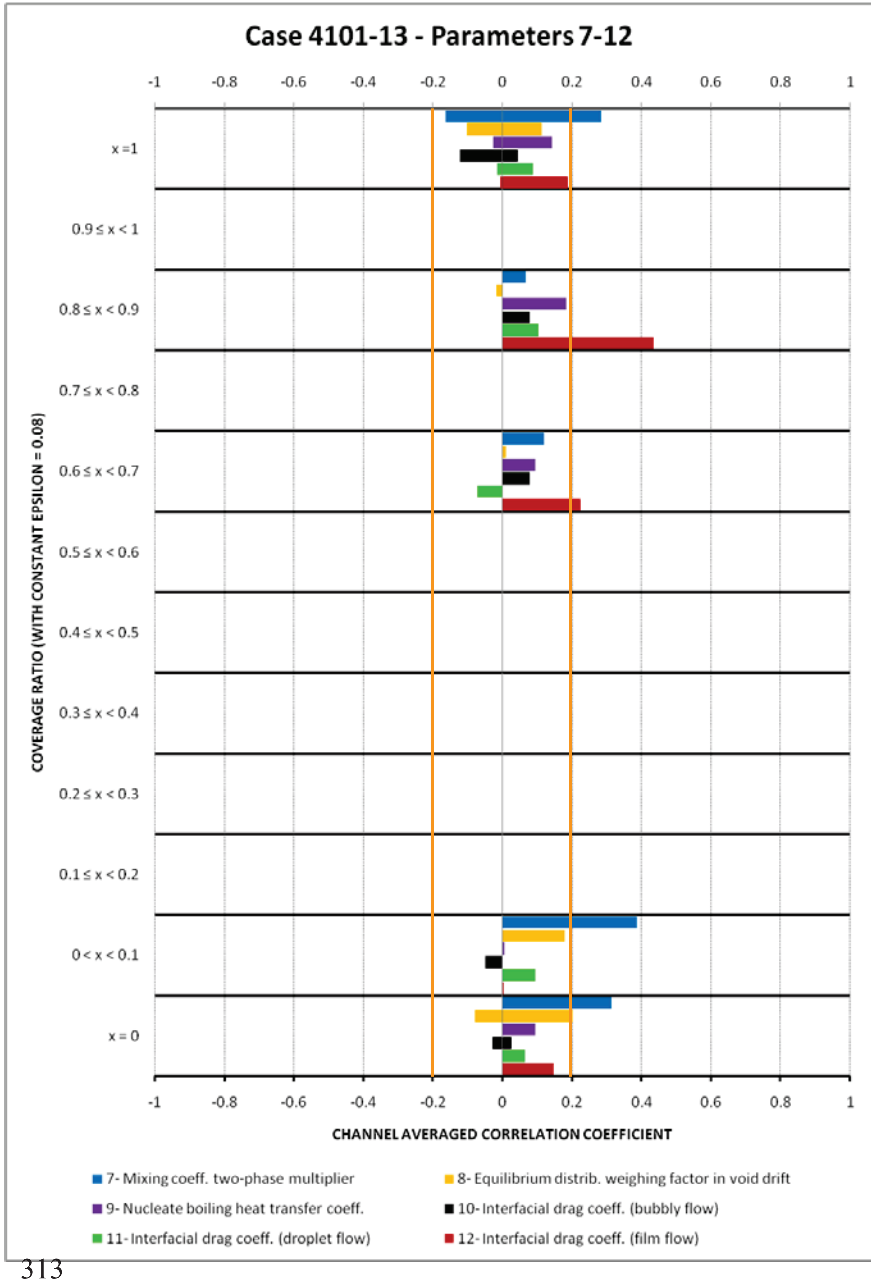
Summary

- The observed discrepancies are believed to be due to two major factors:
    - Previously indicated CTF tendency of void fraction over-prediction;
    - asymmetrical void measurements in regions with otherwise quite symmetrical power load (for example, sub-channels (1, 9), (9, 1), and (9, 9), which are expected to have very close void fractions).
- Nevertheless, it can be seen that the largest deviations are in the corner sub-channels and in the sub-channels connected to the water rod or, in other words, in fluid volumes bounded by unheated walls.

Sensitivity Analysis

Case 4101-13


Case 4101-13 - Parameters 1-6


Case 4101-13 - Parameters 7-12

Sensitivity Analysis

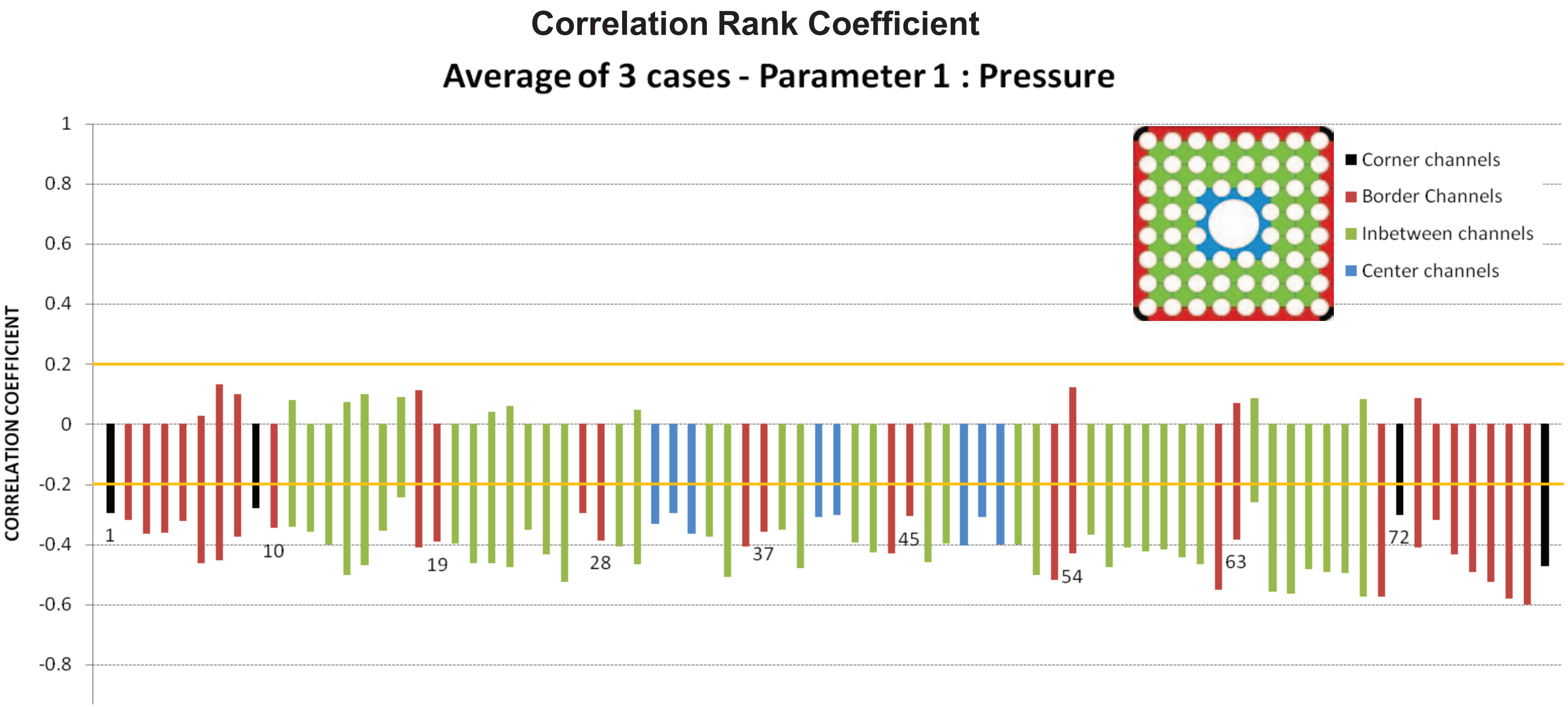
Summary

- The performed sensitivity analyses on the uncertain input parameters led to the following conclusions:
  - From the selected boundary conditions parameters, for all investigated fluid conditions (from bubbly to annular flows), the code predictions are mostly sensitive to the pressure perturbations
  - The uncertainties in the sub-channel area, the only geometry parameter selected for this study, do not significantly affect the code predictions
  - The code predictions highly depend on the variations of three CTF modeling parameters: the single-phase mixing coefficient and its two-phase multiplier (turbulent mixing model), and the interfacial drag coefficients (interfacial friction model)
  - The equilibrium distribution weighing factor in the void drift model does not significantly affect the code void fraction calculations

Sensitivity Analysis

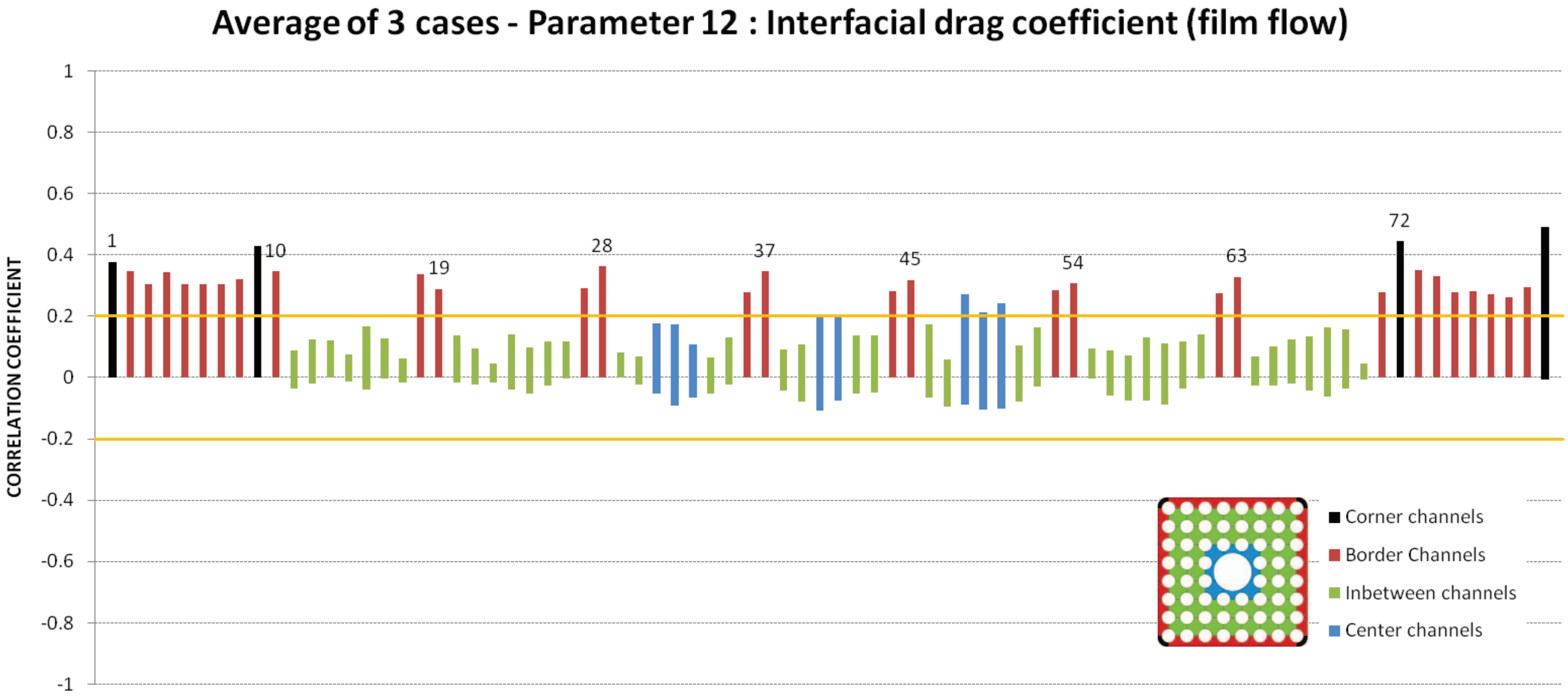
The spatial distribution (over different sub-channels types) of the correlation rank coefficients for the three uncertain input parameters having the major impact on the code sensitivity was also investigated

For each sub-channel, the correlation rank coefficients are averaged over the three test cases (4101-13, 4101-69, and 4101-86)



Sensitivity Analysis

Correlation Rank Coefficient



Sensitivity Analysis

Summary

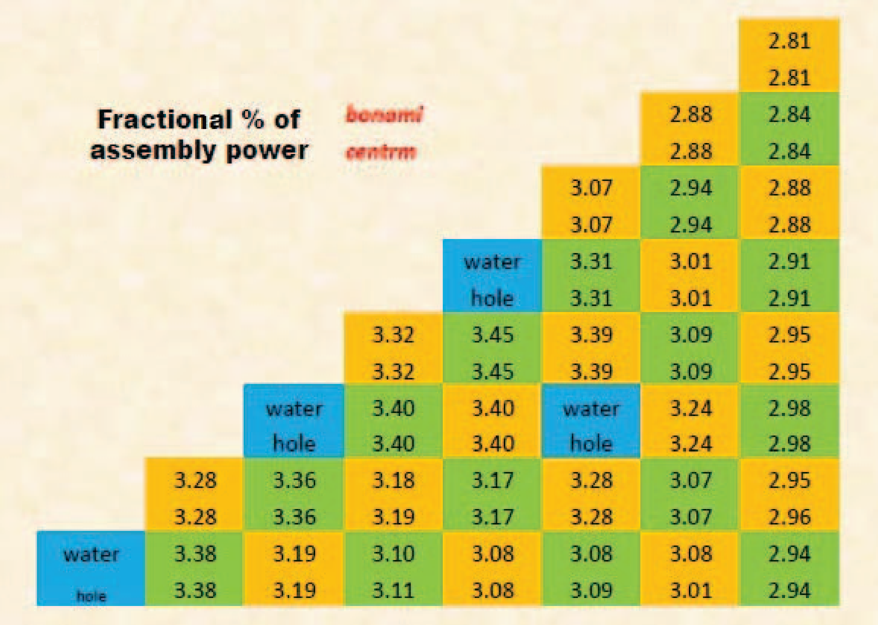
- The spatial distribution (over different sub-channels types) of the correlation rank coefficients have indicated the following:
  - The correlation coefficient for the pressure tends to be equally distributed, while the others two show pronounced dependence on the sub-channel type
  - Variations in the single-phase mixing coefficient affect mostly the code predictions for the interior region of the bundle (internal and central sub-channels)
  - The thermal-hydraulic solution for the fluid volumes connected to unheated walls (corner, side and central sub-channels) has stronger dependence on the liquid film-to-vapor core interfacial drag coefficient

Extension of BEPU to Coupled Codes

- **Propagation of uncertainties in coupled three-dimensional neutronics/thermal-hydraulic calculations involves different input sources of uncertainties and large size of input parameters associated with:**
  - ✓ **Realistic reactor models;**
  - ✓ **Multi-physics phenomena – reactor physics, thermal-hydraulics, and fuel modeling;**
  - ✓ **Non-linear feedback modeling,**
- **Group cross-sections are important input data to coupled core calculations and different approaches have been used for propagation of cross-section uncertainties in core calculations:**
  - ✓ **Statistical sampling at the level of multi-group cross-sections and associated uncertainties (multi-group covariance matrix):**
    - ❖ **All the correlations are taken into account implicitly;**
    - ❖ **This method is called the Random Sampling (XSUSA-SCALE-PARCS).**

Extension of BEPU to Coupled Codes

- **Hybrid approach in which the Generalized Perturbation Theory (GPT) sequence is combined with front end perturbation engine based on statistical sampling:**
  - **TSUNAMI-2D provides a generalized GPT method within SCALE-6.1 to generate covariance matrices for assembly averaged cross-sections;**
  - **The complete covariance matrix (for all cross sections of all fuel assemblies) has to be determined;**
  - **This method is called the Two Step Method (SCALE-DAKOTA-PARCS) – UPC and PSU are exploring this approach**

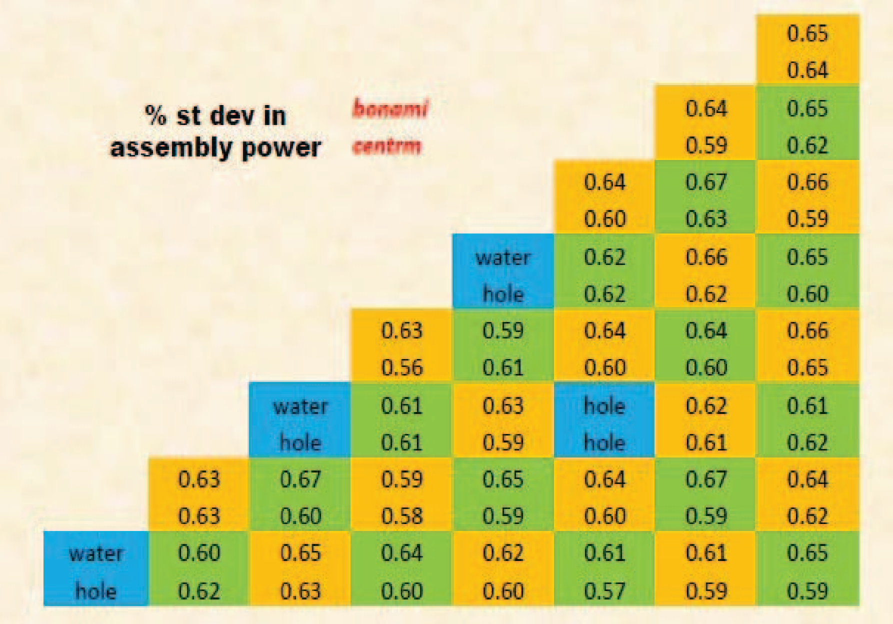


Fractional % of assembly power

bonami
centrm

							2.81 2.81
						2.88 2.88	2.84 2.84
					3.07 3.07	2.94 2.94	2.88 2.88
				water hole	3.31 3.31	3.01 3.01	2.91 2.91
			3.32 3.32	3.45 3.45	3.39 3.39	3.09 3.09	2.95 2.95
		water hole	3.40 3.40	3.40 3.40	water hole	3.24 3.24	2.98 2.98
	3.28 3.28	3.36 3.36	3.18 3.19	3.17 3.17	3.28 3.28	3.07 3.07	2.95 2.96
water hole	3.38 3.38	3.19 3.19	3.10 3.11	3.08 3.08	3.08 3.09	3.08 3.01	2.94 2.94

Mean Values for Statistical Distributions of Relative Pin Powers



% st dev in assembly power

bonami
centrm

							0.65 0.64
						0.64 0.59	0.65 0.62
					0.64 0.60	0.67 0.63	0.66 0.59
				water hole	0.62 0.62	0.66 0.62	0.65 0.60
			0.63 0.56	0.59 0.61	0.64 0.60	0.64 0.60	0.66 0.65
		water hole	0.61 0.61	0.63 0.59	hole hole	0.62 0.61	0.61 0.62
	0.63 0.63	0.67 0.60	0.59 0.58	0.65 0.59	0.64 0.60	0.67 0.59	0.64 0.62
water hole	0.60 0.62	0.65 0.63	0.64 0.60	0.62 0.60	0.61 0.57	0.61 0.59	0.65 0.59

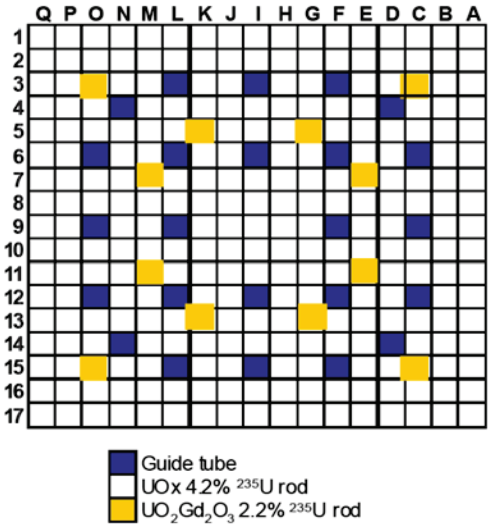
Uncertainties in Pin Powers

Extension of BEPU to Coupled Codes

- **Higher order deterministic method:**
  - The expensive computational cost is one of the important reasons of why only first order derivatives of the response are considered in sensitivity analysis and uncertainty quantification in nuclear reactor calculations;
  - For accurate estimation of sensitivities and uncertainties, nonlinear behavior must be considered, which means higher order derivatives must be determined;
  - The perturbation theory has been developed in order to extend its applicability to estimate higher order variations; however, the computational overhead of higher order perturbation theory is often overwhelming and do not justify the development effort required for their implementation.
- **Feasibility studies**
  - ✓ **Nonlinear ESM-based approach for Hessian matrix construction:**
    - Application of a general reduced order method to constructing second order derivatives of response of interest with respect to all input data (which are often compactly combined in a matrix denoted by the Hessian matrix);
    - Adopting the Efficient Subspace Method (ESM) leads to an optimum implementation in a parallel computing environment.
  - ✓ **Analytic methods for sensitivity analysis of non-linear and transient problems:**
    - Analytic methods for non-linear and transient problems have been developed since the 70's;
    - When implicit methods are utilized to solve the non-linear systems, analytic sensitivity methods can be very efficient;
    - Both direct differentiation and adjoint methods can be used.

Extension of BEPU to Coupled Codes

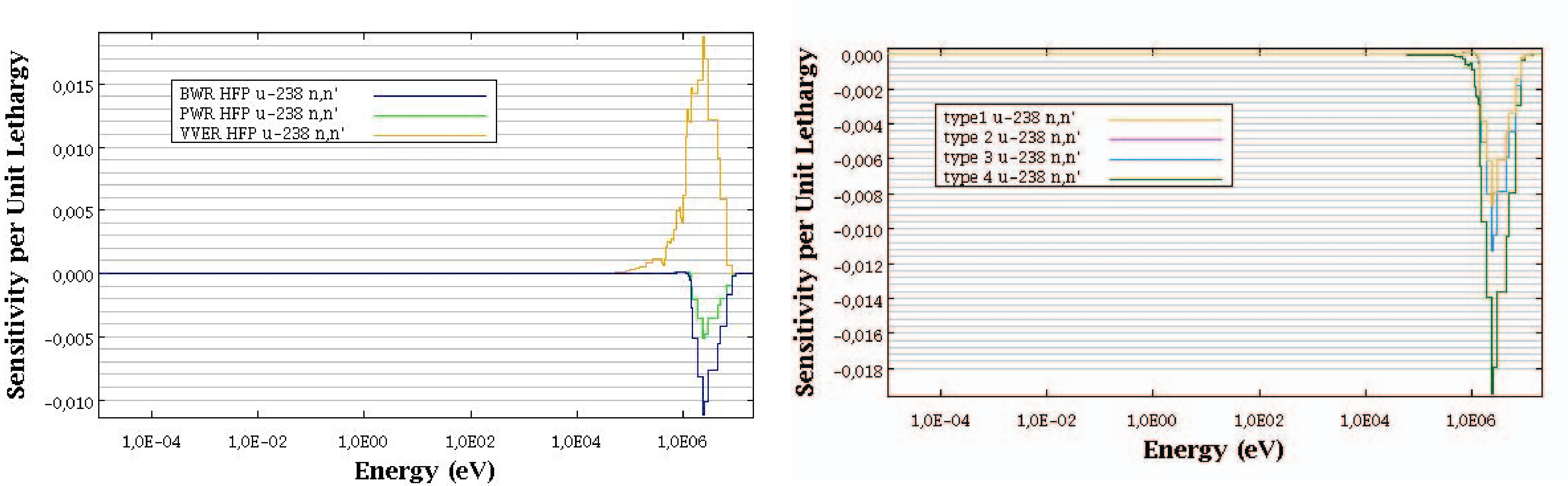
- **The few-group cross-section uncertainties (few-group covariance matrix) are obtained using the SCALE-6.0 44-group covariance matrix as input to the TSUNAMI-2D sequence with GPT in SCALE 6.1**
- **One can define 9-dimensional response vector $R = [\Sigma_{a1}, \Sigma_{a2}, \Sigma_{f1}, \Sigma_{f2}, \nu\Sigma_{f1}, \nu\Sigma_{f2}, D_1, D_2, \Sigma_{1\rightarrow 2}]$ for two-group assembly homogenized cross-sections and obtain a corresponding covariance matrix in which the diagonal elements are the % standard deviations, while off-diagonal elements are the correlation coefficients.**



Reactor Type	Case	k	%Δk/k	Major contributor
BWR	HZP	1.11	0.50	^{238}U (n,γ)
	HFP (40% void)	1.08	0.56	^{238}U (n,γ)
PWR	HZP	1.41	0.46	^{238}U (n,γ)
	HFP	1.39	0.47	^{238}U (n,γ)
VVER	HZP	1.32	0.77	^{238}U (n,n')
	HFP	1.31	0.78	^{238}U (n,n')
GEN III	Type 1 (UOX 4,2% ^{235}U)	1.25	0.49	^{238}U (n,γ)
	Type 2 (UOX 4,2% ^{235}U+ $UO_2Gd_2O_3$ 2,2% ^{235}U)	1.12	0.49	^{238}U (n,γ)
	Type 3 (UOX 3,2% ^{235}U+ $UO_2Gd_2O_3$ 1,9% ^{235}U)	0.96	0.53	^{238}U (n,γ)
	Type 4 (MOX)	1.07	0.97	^{238}U (n,n')

Extension of BEPU to Coupled Codes

Reactor Type	Case	Responses (%ΔR/R)								
		Σ_{c1}	Σ_{c2}	Σ_{f1}	Σ_{f2}	$\nu\Sigma_{f1}$	$\nu\Sigma_{f2}$	D_1	D_2	$\Sigma_{1\to2}$
BWR	HZP	1.1975	0.5127	0.6811	0.3235	0.9783	0.4485	0.8435	0.1342	1.0070
	HFP (40% void)	1.2880	0.5458	0.7318	0.3246	1.0131	0.4494	0.9163	0.1593	1.2255
PWR	HZP	1.2870	0.8175	0.3558	0.3167	0.5121	0.4436	0.8797	0.1505	1.2065
	HFP	1.2869	0.8170	0.3569	0.3176	0.5128	0.4443	0.8815	0.1495	1.2068
VVER	HZP	1.2462	0.6275	0.7141	0.3130	1.0483	0.4408	0.9175	0.1243	0.1648
	HFP	1.2503	0.6322	0.7193	0.3134	1.0541	0.4411	0.9176	0.1225	1.1680
GEN III	Type 1 (UOX 4,2% ^{235}U)	1.3013	0.6218	0.3706	0.3233	0.5708	0.4484	0.9082	0.1543	1.2487
	Type 2 (UOX 4,2% ^{235}U+ $UO_2Gd_2O_3$ 2,2% ^{235}U)	1.2882	0.4588	0.3751	0.3237	0.5820	0.4487	0.9068	0.1534	0.9068
	Type 3 (UOX 3,2% ^{235}U+ $UO_2Gd_2O_3$ 1,9% ^{235}U)	1.2748	0.3716	0.4994	0.3295	0.7626	0.4529	0.9041	0.1535	1.2429
	Type 4 (MOX)	1.3975	0.7524	0.4449	0.6261	0.7748	1.0863	0.9764	0.1600	1.4730



Response sensitivity to U-238 n,n'

Extension of BEPU to Coupled Codes

The obtained results for different LWR types and cases with GPT in SCALE-6.1 indicated the following tendencies:

- Group 1 (fast) cross-section uncertainty is larger than Group 2 (thermal) cross-sections uncertainty;
- Uncertainty contributions:
  - A major contributor to Group 1 (fast) cross-sections is U-238 inelastic scattering and U-238 capture;
  - U-238 inelastic scattering uncertainty is quite large;
  - 40% void (and higher) exhibit larger uncertainty in k_∞ due to harder flux spectrum.
- Uncertainty (correlation) contribution:
  - U-238 inelastic scattering uncertainty is quite large, and dominates correlation coefficient.

Extension of BEPU to Coupled Codes

- The current state-of-art coupling in reactor safety analysis performed with coupled neutronics/thermal-hydraulic calculations is based on an explicit time integration method which works well when power is not changing rapidly or when detailed spatial feedback is not important
- However, for analysis of transients such as BWR stability events, where fast power response and detailed spatial kinetics models are important, the time step size is limited by the error with the explicit treatment of the feedback parameters
- To overcome this limitation in this type of analysis a fully implicit coupling of neutronics and thermal-hydraulics codes was developed at PSU and implemented into the coupled TRACE/PARCS code package
- The fully implicit coupling involves forming the full Jacobian for all physics involved in the analysis and solving the entire nonlinear system.

Extension of BEPU to Coupled Codes

- **Sensitivity analysis and uncertainty quantification of tightly coupled reactor physics thermal-hydraulic analyses require the development of not only first but also higher (at least second) order sensitivities.**
- **Analytic sensitivity methods (direct differentiation and adjoint methods) can be used in conjunction with an implicit solution approach to develop numerically efficient methods for addressing the issue of high dimensionality in uncertainty quantification in reactor simulations.**
- **Experience from other areas can be utilized - analytic methods for sensitivity analysis of non-linear and transient problems have been developed since the 70's**
- **When implicit methods are utilized to solve the non-linear systems, analytic sensitivity methods can be very efficient.**

Extension of BEPU to Coupled Codes

- A non-linear system can be denoted as a residual vector R(u) = 0, where u is the unknown system response. Using the Newton-Raphson method, the system is iteratively computed as:

(1)
$$\mathbf{u}^{I+1} = \mathbf{u}^{I} - \left[\frac{d\mathbf{R}}{d\mathbf{u}}(\mathbf{u}^{I})\right]^{-1} \mathbf{R}(\mathbf{u}^{I})$$

where *d*R/*d*u is the Jacobian.

- For large systems, more than 90% of the computational time is devoted in computing and evaluating the Jacobian.
- In a sensitivity analysis, the response function f (x) = f (u(x), x) is expressed as a function of the system response u and input state x. Differentiating f with respect to each input state component x_i results

(2)
$$\frac{d\mathbf{f}}{dx_i} = \frac{\partial \mathbf{f}}{\partial \mathbf{u}}\frac{d\mathbf{u}}{dx_i} + \frac{\partial \mathbf{f}}{\partial x_i}$$

Extension of BEPU to Coupled Codes

- Using the direct differentiation approach, the system response sensitivity du/dx_i is computed by differentiating the residual equation:

(3) $$\frac{d\mathbf{R}(\mathbf{u}(\mathbf{x}),\mathbf{x})}{dx_i} = \frac{\partial \mathbf{R}}{\partial \mathbf{u}}\frac{d\mathbf{u}}{dx_i} + \frac{\partial \mathbf{R}}{\partial x_i} = \mathbf{0} \Rightarrow \frac{d\mathbf{u}}{d\mathbf{x}_i} = -\left[\frac{\partial \mathbf{R}}{\partial \mathbf{u}}\right]^{-1}\frac{\partial \mathbf{R}}{\partial \mathbf{x}_i}$$

- The sensitivity calculation of the above equation is performed after the iterative solution of equation (1) using the same Jacobian:
  - Thus, it only requires one back-substitution per x_i
  - Typically, 1% additional computational time is required per state variable x_i
  - The total computational overhead is proportional to the number of state variables p (rank of x).

Extension of BEPU to Coupled Codes

- **The adjoint sensitivity analysis method can be derived by augmenting the functional response function f = f (u(x), x) + λ^TR by a Lagrange multiplier λ noting that R = 0, which after differentiation results to**

(4)

$$\frac{df_k}{dx_i} = \frac{\partial f_j}{\partial \mathrm{u}}\frac{d\mathrm{u}}{dx_i} + \frac{\partial f_k}{\partial x_i} + \lambda^T\left[\frac{\partial \mathrm{R}}{\partial \mathrm{u}}\frac{d\mathrm{u}}{dx_i} + \frac{\partial \mathrm{R}}{\partial x_i}\right] = \frac{\partial f_k}{\partial x_i} + \lambda^T\frac{\partial \mathrm{R}}{\partial x_i} \quad \text{with} \quad \lambda = -\left[\frac{\partial \mathrm{R}}{\partial \mathrm{u}}\right]^{-T}\left(\frac{\partial f_k}{\partial \mathrm{u}}\right)^T$$

  - **The adjoint sensitivity calculation of the above equation also uses the same Jacobian as the analysis.**
  - **The total computational overhead is proportional to the number of response functionals *q* (rank of f).**
  - **Second order sensitivities can be obtained by a hybrid direct differentiation-adjoin method that results into *p* + *q* additional back substitutions.**

- **In summary, sensitivity formulations for fully coupled kinetic-thermal-hydraulic implicit analyses are being developed and to be implemented in the TRACE/PARCS code package.**

- **Following this implementation will be the demonstration of sensitivity analysis in the uncertainty quantification of power level, power distribution, void distribution, pressure, etc of OECD/NRC Peach Bottom Turbine Trip benchmark simulations.**

Conclusions

- Extension of BEPU to sub-channel codes is demonstrated with COBRA-TF for steady-state void fraction prediction in a BWR bundle using SUSA
- Similar studies with COBRA-TF and SUSA are underway for:
  - DNB predictions and transient applications;
  - PWR bundles using the OECD/NRC PSBT benchmark database;
  - Exercise II-3 of the OECD LWR UAM benchmark.
- The two-group homogenized assembly cross-section uncertainties have been generated with SCALE-6.1, and UPC and PSU are planning to apply the hybrid two-step method to propagate these uncertainties into core calculations as part of Exercise I-3 of the OECD LWR UAM benchmark
- Feasibility studies are also underway for implementing and using higher order deterministic methods in TRACE/PARCS calculations with fully implicit coupling
- These methods are designed to explore the full phase space of input parameters and to take the non-linearity of the feedback mechanisms into account in the multi-physics calculations within the framework of Exercise III-3 of OECD LWR UAM benchmark

Application of the Integrated Safety Assessment Methodology to Sequences with Loss of Component Cooling Water System

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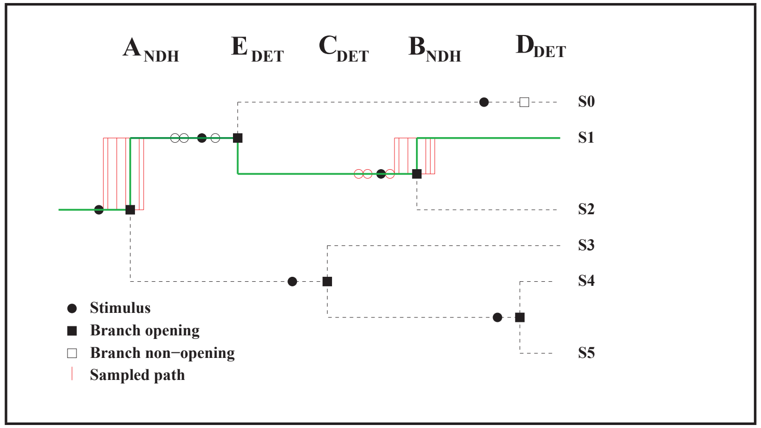
OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations
Barcelona, Spain, November 16-18, 2011

1. INTRODUCTION

- The ISA methodology has been developed in the Modeling and Simulation (MOSI) branch of Consejo de Seguridad Nuclear.

  The numerical result of this methodology consists of the exceedance frequency of damage for the sequences stemmed from an initiating event.
- This methodology aims:
  - Automatic delineation of dynamic event trees (dynamic PSA),
  - to take into account the uncertainties of the sequences. Uncertainties with time or parameter dependencies like: human actions (time), break area, thermal power, pressures, mass flows, stochastic phenomena...
- In this application the analysis has been focused on uncertain times since they are expected to be dominant in this kind of sequences (non deterministic headers),



1. INTRODUCTION. ISA methodology

- In PSA event trees there are two options for a header: success or failure.
- **In ISA methodology there are three options for a header: demanded with success, demanded with failure and not demanded.**
- In PSA event trees there are two final states for a sequence: success or damage with damage probability 0 or 1.
- **In ISA methodology the final state of a sequence is a random variable that depends on the performance of headers with time uncertainty: damage with probability P_D and success with probability P_S which fulfil that $P_D + P_S = 1$.**

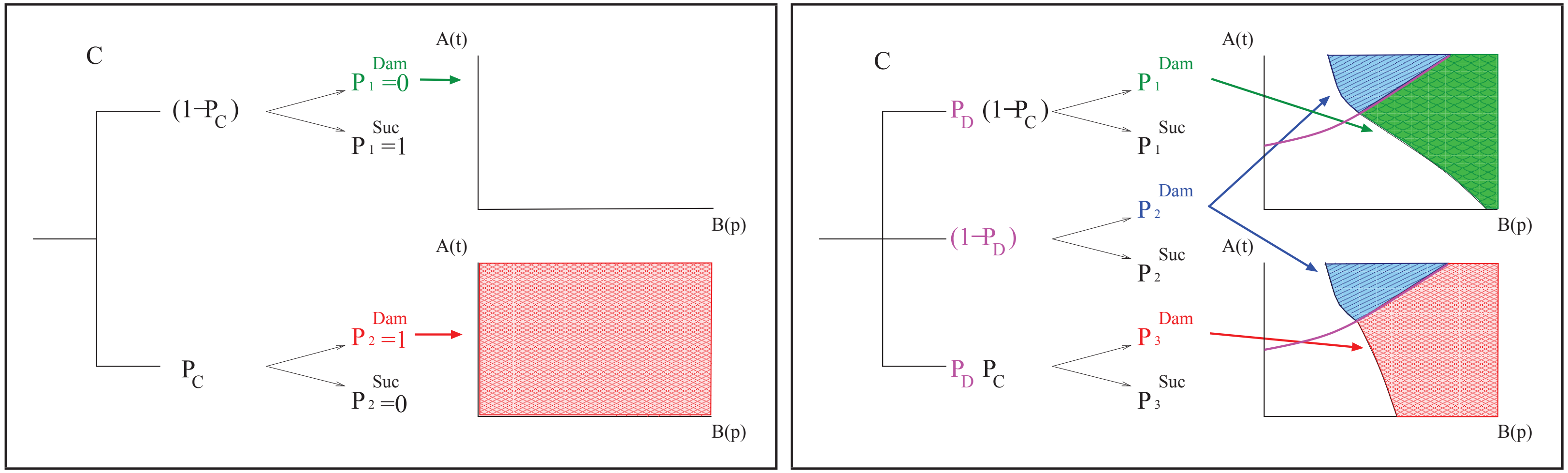


Figure 1: ISA Methodology general diagram

1. INTRODUCTION. ISA methodology

- ○ In PSA a human action is failed if it is not performed on time (available time).
- **In ISA methodology a human action never is failed only is delayed (the integral of the density probability function is the unity).**
- ○ In PSA the systems failure probabilities do not depend on the dynamics of the sequence.
- **In ISA methodology the failure rate could depend on process variables (i.e. temperature or humidity conditions in the proximity of a pump).**

  The general equations for the damage frequency are the Theory of Stimulated Dynamics (TSD). These equations are simplified if the failure rates do not depend on the process variables. **This is the case in SM2A Project.**

1. INTRODUCTION. ISA methodology

The main steps are (simplified version):

Step 1 First, a Generic Event Tree (GET) is obtained from previous PSA.
The objective of this step is only to identify the headers.

Step 2 **Sequence Generation Module**: A Dynamic Event Tree (DET), without time delays, is performed with the previous headers.
In general the DET and the GET are different.

Step 3 **Transition from Sequence Generation Module to Path Analysis Module.**
With the information obtained from the DET is it possible to obtain which sequences with time uncertainty (U sequences) have a damage domain (the region where the safety limit is exceeded).

Step 4 **Path Analysis Module.**
Obtention of the Damage Domain in the sequences of interest (sequences with damage probability different from 0 and 1).

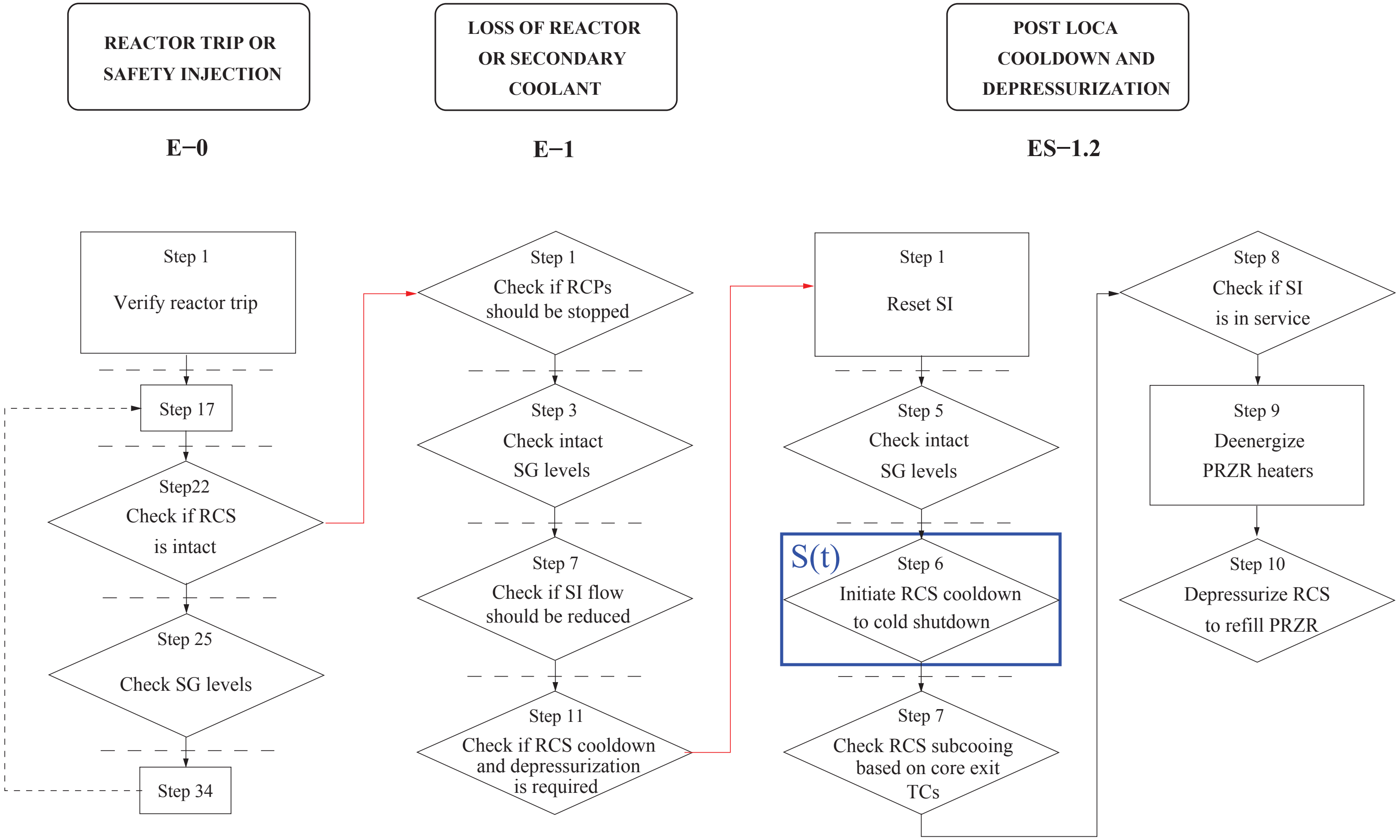
Step 5 **Risk Assessment Module.**
In this step the Theory of Stimulated Dynamics (TSD) equations are integrated inside the Damage Domain.
The result of this final step is the Damage Exceedence Frequency.

2. SM2A Project. Sequences with loss of CCW

- The OECD/NEA Committee for Safety of Nuclear Installations (CSNI) has promoted some international initiatives to develop suitable criteria for determining the "safety margins" to provide an acceptable level of safety in NPP. With this purpose, the Task Group on SM2A has been set up to carry out a pilot application project of calculating the increase in exceedance frequency of damage resulting from a power uprate of 10% in Zion NPP. Participants: EDF, PSI, CNSNS, STUK, KAERI/KINS, JNES, NRC and CSN. **See paper Safety Margin Assessment (SM2A): Stimulation for further development of BEPU approaches. M. Zimmermann (PSI).**
- Spanish Participant: Consejo de Seguridad Nuclear in collaboration with Universidad Politécnica de Madrid and Indizen.
- Objective: Increase of damage exceedence frequency in a power uprate from 100% to 110%.
- CSN-UPM Group selected the sequences with loss of Component Cooling Water (CCW) System.
- The simulations were performed with two models of Zion NPP (MAAP and TRACE codes).
- **Uncertainties**:
  - Instant of the occurrence of seal LOCA (SLOCA). **SLOCA(t)**
  - Beginning of secondary side depressurization to cool (55 K/h) and depressurize primary side after seal LOCA. **S(t)**
  - Recovery time of CCW system. **R(t)**

In this presentation only the results from TRACE are showed (about 80 simulations) (MAAP analysis is included in SM2A final report).

2. Loss of CCW/SW with SLOCA. Human actions in a PWR Westinghouse design



REACTOR TRIP OR SAFETY INJECTION

E−0

Step 1
Verify reactor trip

Step 17

Step22
Check if RCS
is intact

Step 25
Check SG levels

Step 34

LOSS OF REACTOR OR SECONDARY COOLANT

E−1

Step 1
Check if RCPs
should be stopped

Step 3
Check intact
SG levels

Step 7
Check if SI flow
should be reduced

Step 11
Check if RCS cooldown
and depressurization
is required

POST LOCA COOLDOWN AND DEPRESSURIZATION

ES−1.2

Step 1
Reset SI

Step 5
Check intact
SG levels

S(t)

Step 6
Initiate RCS cooldown
to cold shutdown

Step 7
Check RCS subcooing
based on core exit
TCs

Step 8
Check if SI
is in service

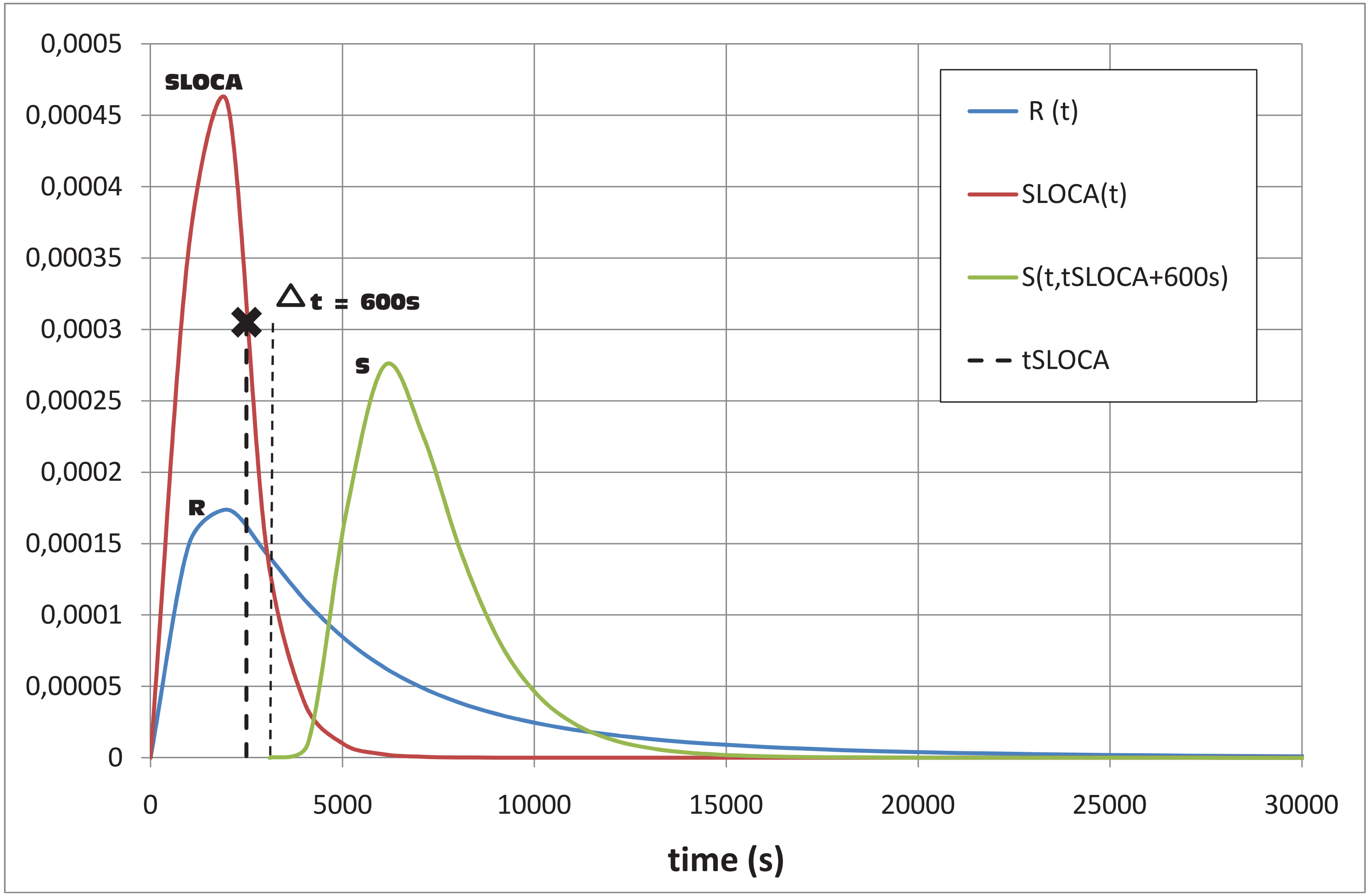
Step 9
Deenergize
PRZR heaters

Step 10
Depressurize RCS
to refill PRZR

Figure 2: Emergency operating procedures involved in a Seal LOCA for PWR Westinghouse design

2. SM2A Project. Sequences with loss of CCW

Probability Density Functions of time delays of human actions and stochastic phenomena.



2-A. SEQUENCE GENERATION. Loss of CCW/SW with SLOCA

A Generic Event Tree is obtained from previous PSA,

Header	Description
$SLOCA$	RCP seals failure
L_R	Recovery of CCW/SW and LPSI(1/2) available (including recirculation phase)
H_R	Recovery of CCW/SW and HPSI(2/4) available
S	Beginning of primary cooling at 55K/h
A	Accumulators Discharge (4/4)

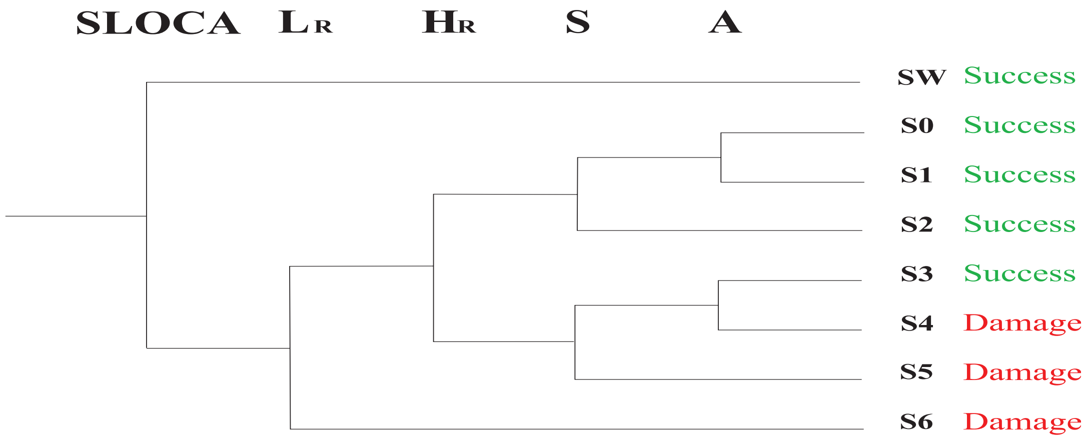
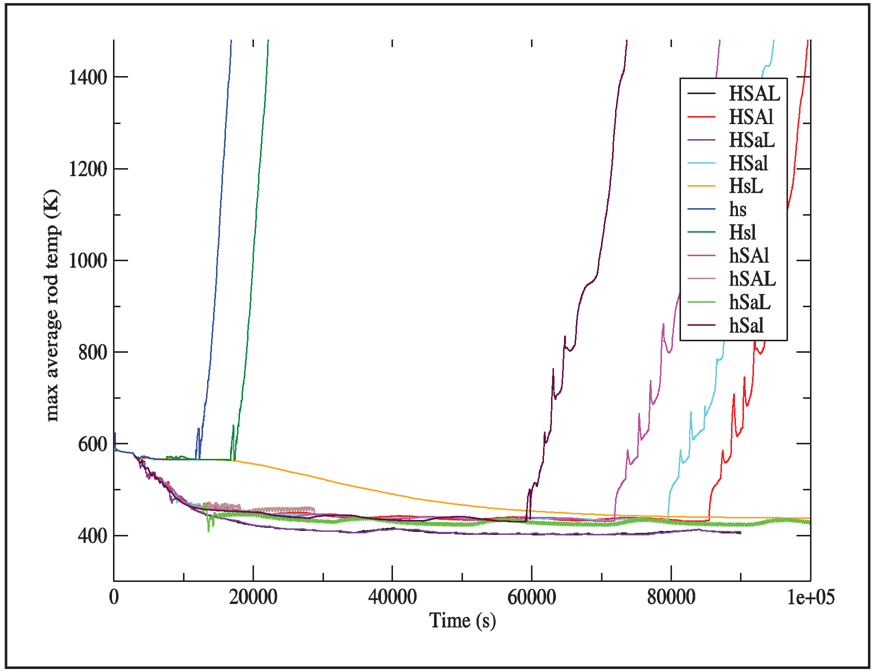


Figure 3: Loss of CCW/SW. Generic Event Tree.

2-A. SEQUENCE GENERATION. Dynamic Event Tree (DET). Loss of CCW/SW with SLOCA. TRACE Code

A Dynamic Event Tree is performed without time delays.



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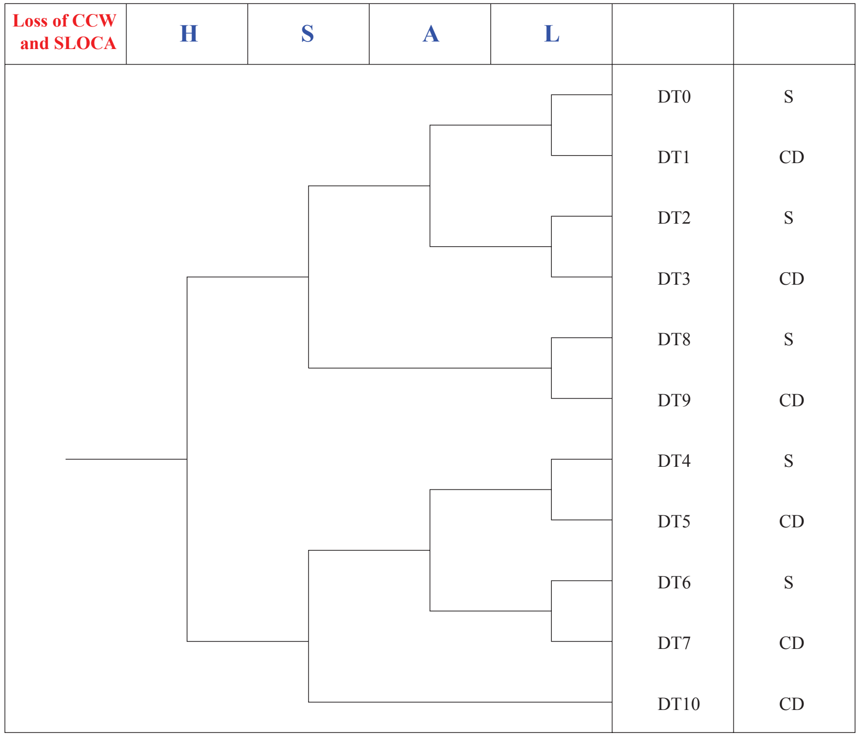


Figure 4: Dynamic Event Tree (DET). Maximum cladding temperature. Loss of CCW/SW with a SLOCA.

2-A. SEQUENCE GENERATION. Dynamic Event Tree (DET). Loss of CCW/SW with SLOCA. TRACE Code

DET	Dynamic sequence	Time of H header	Time of S header	Time of A header	Time of L header	PCT
DT0	HSAL	2619	3100	5773	7400, R	622 K
DT1	HSAl	2619	3100	5773	(7400, R)	DAMAGE
DT2	HSaL	2619	3100	(5773)	7400, R	622 K
DT3	HSal	2619	3100	(5773)	(7400, R)	DAMAGE
DT4	hSAL	(2619)	3100	5395	11152	622 K
DT5	hSAl	(2619)	3100	5395	(11152)	DAMAGE
DT6	hSaL	(2619)	3100	(5395)	10450	622 K
DT7	hSal	(2619)	3100	(5395)	(10450)	DAMAGE
DT8	HsL	2619	(3100)	–	7400, R	622 K
DT9	Hsl	2619	(3100)	–	(7400, R)	DAMAGE
DT10	hs	(2619)	(3100)	–	–	DAMAGE

The brackets (t_0) mean that the system has been demanded in t_0 but it has been failed on demand or was unavailable.

2-A. SEQUENCE GENERATION. Dynamic Event Tree (DET). Loss of CCW/SW with SLOCA. MAAP / TRACE Codes

MAAP DETs are performed in an automatic way coupling SCAIS (which includes SIMPROC module) with MAAP.

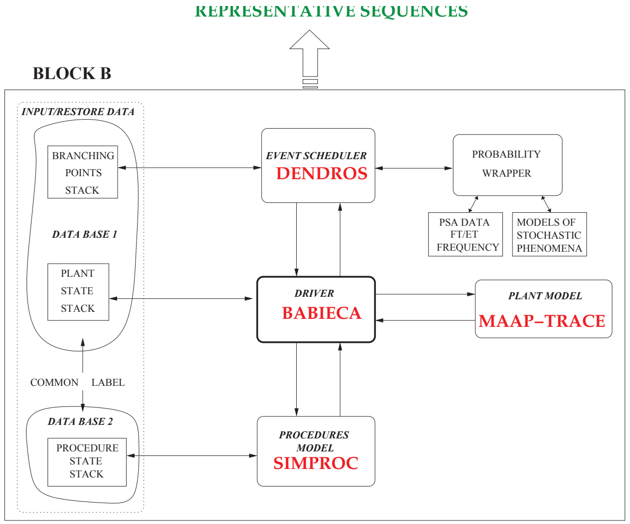


Figure 5: Scheme of Path and Sequence Generation module

This stage of ISA methodology is performed in a similar way to MCDET Methodology (See paper of J. Peschke).

TRACE At present the coupling of TRACE with SCAIS has been obtained but the automatic delineation of DET with TRACE is still on progress.

2-A. SEQUENCE GENERATION.
Loss of CCW/SW with SLOCA. TRACE Code

Taking into account the results from DET it is possible to obtain the final status of each sequence with time uncertainty (U sequences):

Sequence with time uncertainty	DET (TRACE)	Final status (TRACE)
$U0$ (H_RSAL_R)	DT0, DT5 DT8 DT10	S,D,S,D ⇒ DD
$U1$ (H_RSaL_R)	DT2, DT7, DT8, DT10	S,D,S,D ⇒ DD
$U2$ (H_RsAL_R)	DT8, DT10	S,D ⇒ DD
$U3$ $(hSAL_R)$	DT4, DT5, DT10	S,D,D ⇒ DD
$U4$ $(hSaL_R)$	DT6, DT7, DT10	S,D,D ⇒ DD
$U5$ (hs)	**DT10**	D ⇒ D
$U6$ (l)	DT1, DT3, DT5, DT7, DT9	D,D, D,D, D,D ⇒ D

Table 1: Finding the final status of each sequence with time uncertainty (U sequences). TRACE code

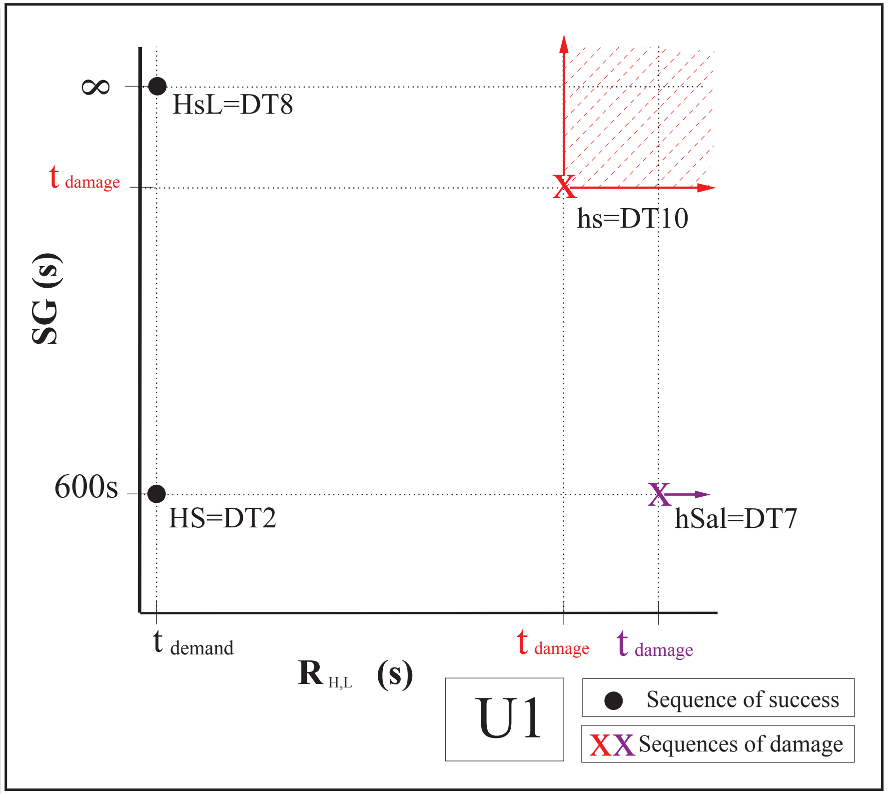


Figure 6: Finding the damage domain of sequence $U1$.

2-A. SEQUENCE GENERATION. Generic Event Tree with Uncertainty. Loss of CCW/SW with SLOCA. TRACE Code

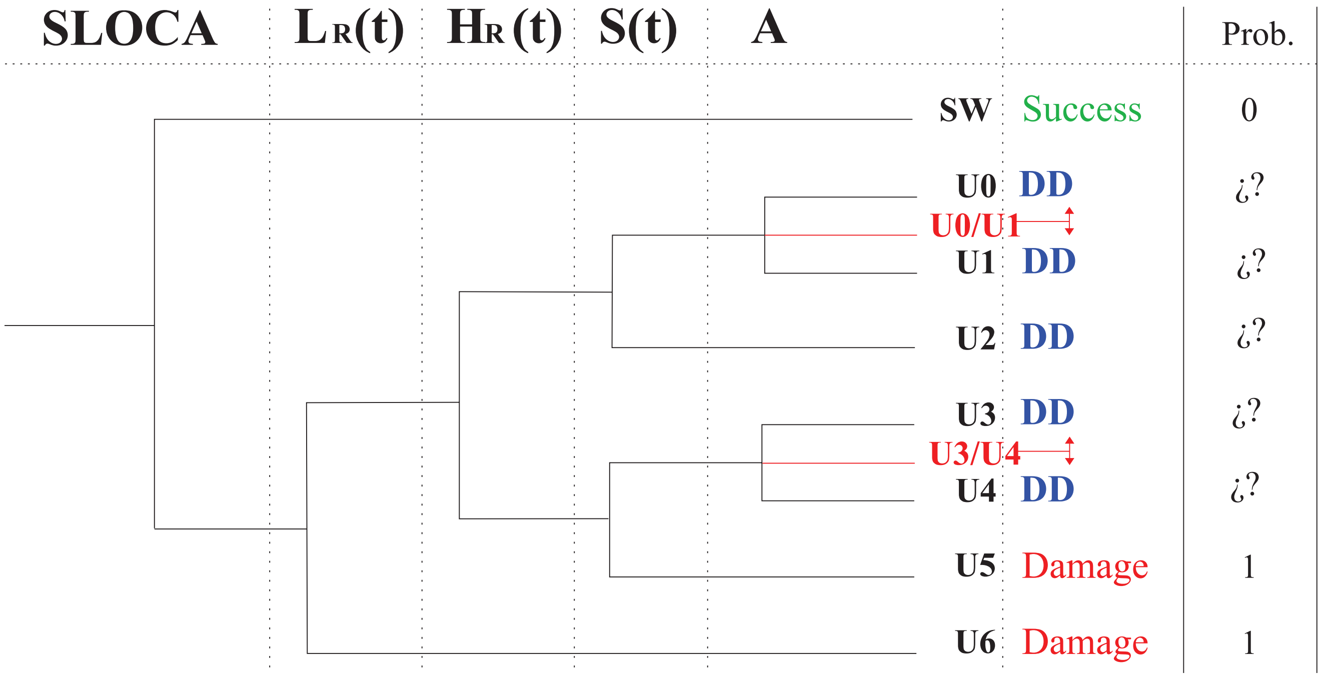


Figure 7: Generic Event Tree with Uncertainty. Loss of CCW/SW with SLOCA.

Only it is necessary to obtain the Damage Domain of U0 and U1 because the others sequences with Damage Domain (U2, U3 and U4) have a frequency lower than the frequency threshold of SM2A exercise ($f < 10^{-7}y^{-1}$).

2-B. PATH ANALYSIS. Loss of CCW/SW with SLOCA. TRACE code.

Seeking of Damage Domain. Example: sequence U1 ($H_R S a L_R$).

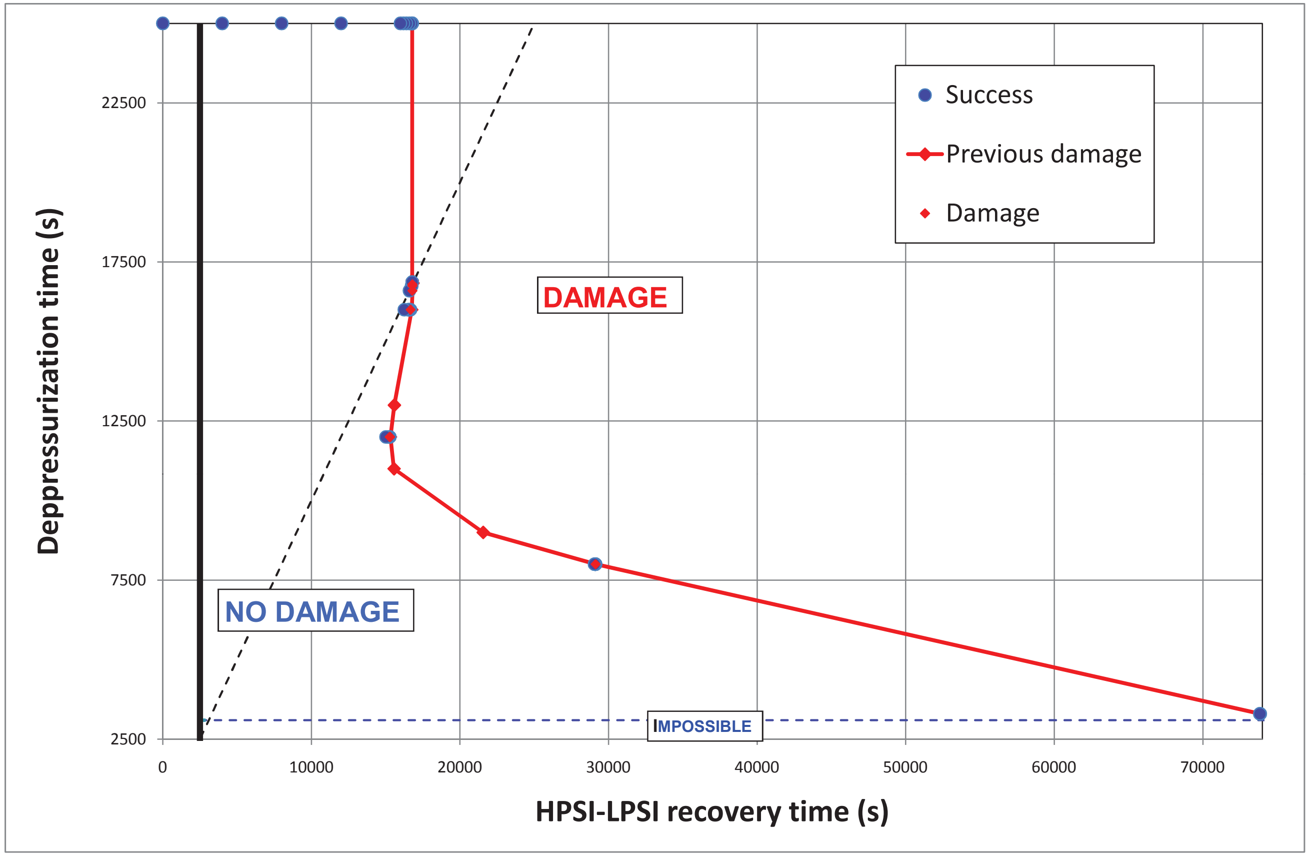


Figure 8: Seeking of damage domain of sequence HSaL (U1). Simulations with TRACE code.

2-B. PATH ANALYSIS. Loss of CCW/SW with SLOCA. TRACE code.

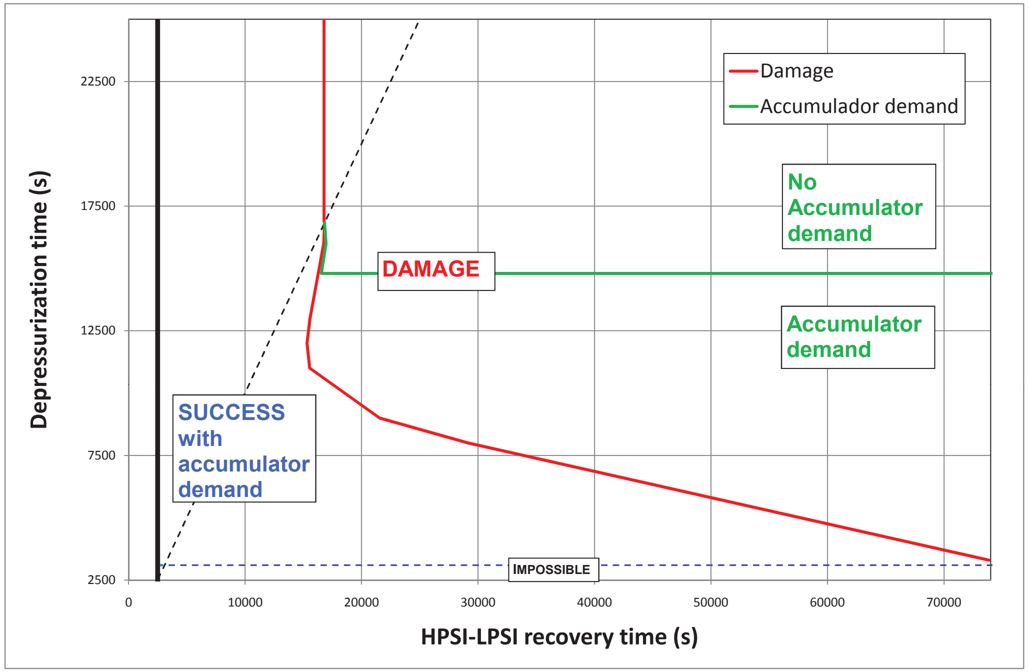


Figure 9: Damage domain of sequence HSaL (U1). Simulations with TRACE code.

The available time of recovering the CCW System is a function of the beginning of secondary side depressurization and ACC status (**in classical PSA this available time has a value but not a function**).

$$t_{U1}^{\text{REC}} = f_1 \left(t^{\text{SG}} | \text{ACC unavailable} \right)$$

An unexpected bifurcation has been obtained. There are sequences with and without accumulators demand inside the domain.

2-B. PATH ANALYSIS. Loss of CCW/SW with SLOCA. TRACE code.

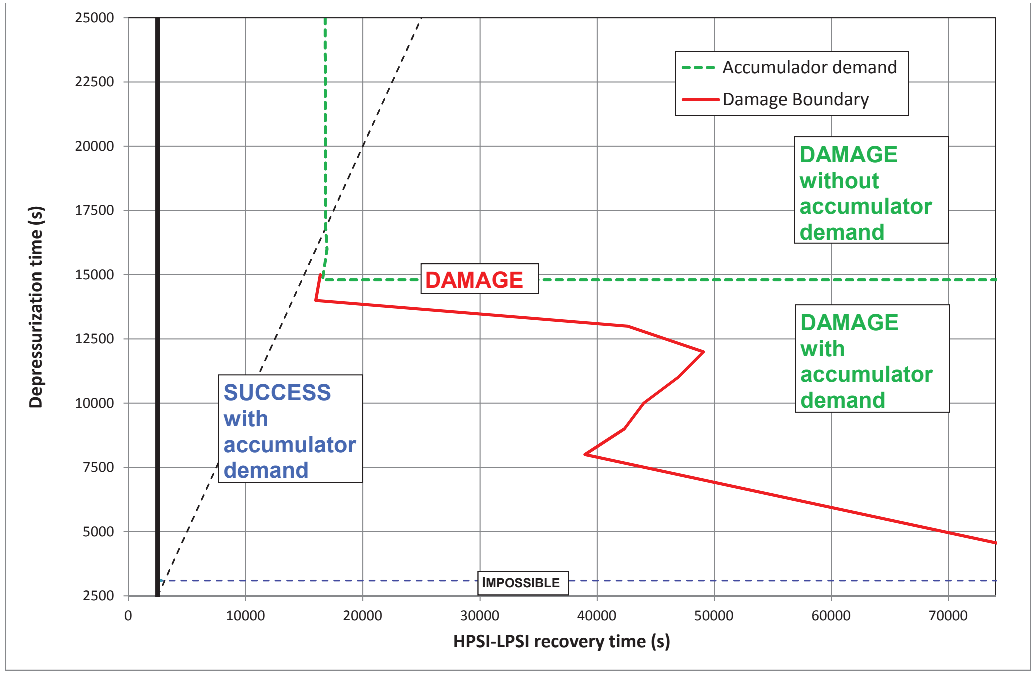


Figure 10: Damage domain of sequence HSAL (U0). Simulations with TRACE code.

Now, there is a different function for the available time of recovering the CCW System.

$$t_{U0}^{\mathsf{REC}} = f_0\left(t^{\mathsf{SG}} | \mathsf{ACC\ available}\right)$$

2-B. PATH ANALYSIS. Loss of CCW/SW with SLOCA. TRACE code.

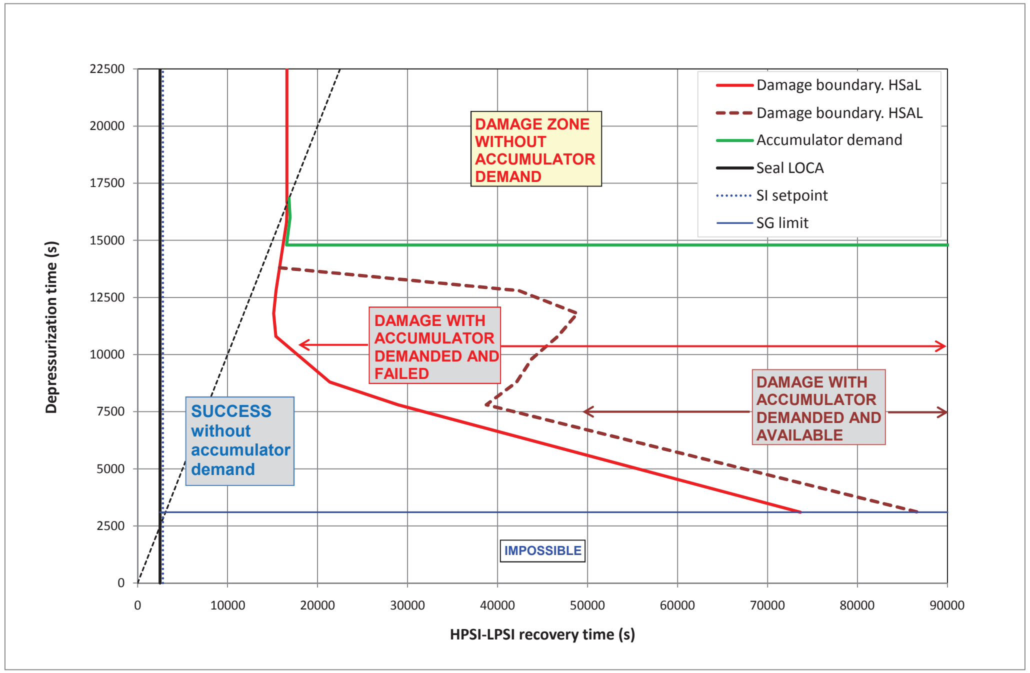


Figure 11: Comparison of the damage domain obtained with TRACE for the sequences HSAL and HSaL

$$t_{U0}^{\text{REC}}(t^{SG}) > t_{U1}^{\text{REC}}(t^{SG})\ \forall t^{SG}$$

Now, the Probability Density Functions (pdf) of time delays and stochastic phenomena (Seal LOCA) are taking into account.

2-C. PROBABILITY AND FREQUENCY DATA.
Loss of CCW/SW with SLOCA.

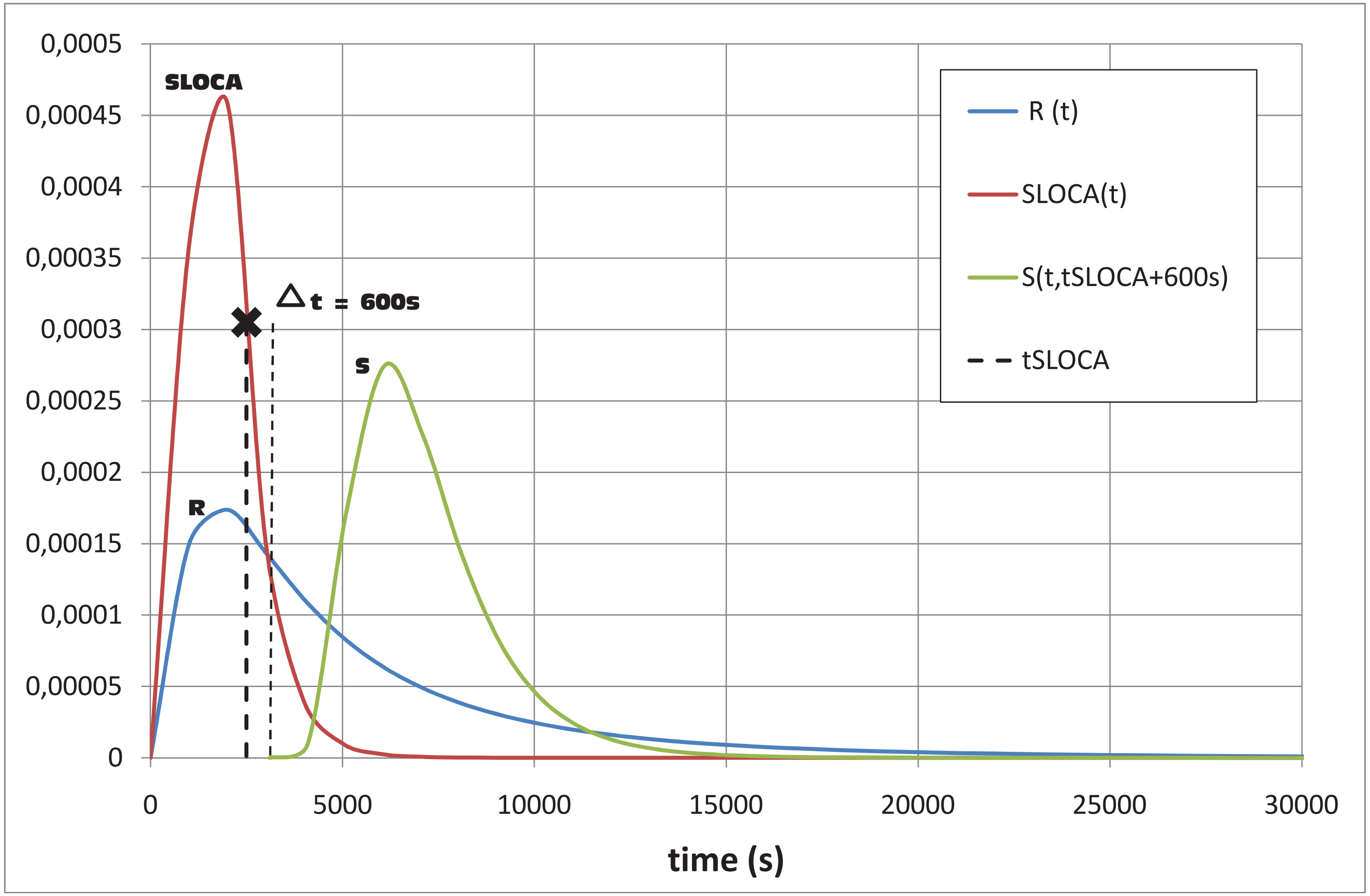
- Pre-existent PSA's, stochastic phenomena models and operator procedures are the sources for the obtaining of the probabilities (for deterministic and stochastic headers) to be used by the rest of the blocks.

System failure probability

Initiator	Frequency (y^{-1})
$Loss\ of\ CCW/SW$	$2 \cdot 10^{-3}$

Header	Type of probability	Failure probability	Distribution function
$SLOCA$	Stochastic	$2.1 \cdot 10^{-1}$	Lognormal $(\mu = 7.4955, \sigma = 0.4214)$
$Recovery$	Stochastic	0	Lognormal $(\mu = 8.2319, \sigma = 0.8690)$
H	Deterministic	$2.2 \cdot 10^{-5}$	-
S	Stochastic	0	Lognormal $(\mu = 8.2091, \sigma = 0.4338)$
A	Deterministic	$9.4 \cdot 10^{-4}$	-
L	Deterministic	$5.6 \cdot 10^{-5}$	-

2-C. PROBABILITY AND FREQUENCY DATA.
Loss of CCW/SW with SLOCA. Probability Density Functions



2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA.

We perform the Risk Assessment in 3 steps:

2NDH **Case with two uncertain times, S(t) and R(t). SLOCA happens at a fixed time,** $t = t_0 = 2500s$**.**

3NDH **Case with three uncertain times, S(t), R(t) and SLOCA(t).**

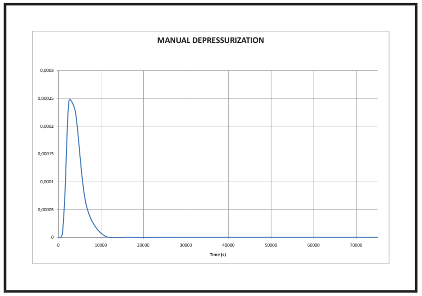
10UP **10% Power Uprate.**

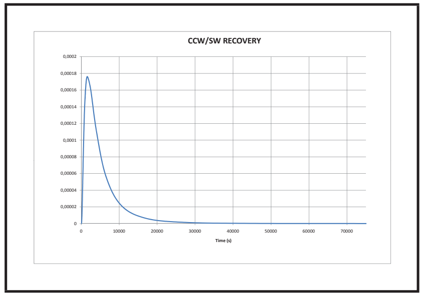
2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Step 1: 2NDH

Case 2NDH in which SLOCA happens at $t = t_0 = 2500s$. Example, Sequence U0, accumulator demanded and available:

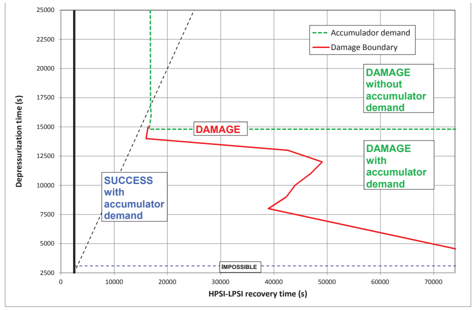
$$\phi_{\text{dam}} = \underbrace{\phi_{\text{ini}} \cdot P_{SLOCA} \cdot (1 - P_H) \cdot (1 - P_L) \cdot (1 - P_A)}_{\text{U0 Sequence Frequency}} \cdot \underbrace{\sum_{p=1}^{DD_{U0}} \left(h_p^S \cdot h_p^{\mathrm{R}} \Delta\tau_S \Delta\tau_R \right)}_{\text{Damage Probability of U0 sequence}}$$

Where $h^S(t - t_{SLOCA})$ and $h^{\mathrm{R}}(t)$ are known:



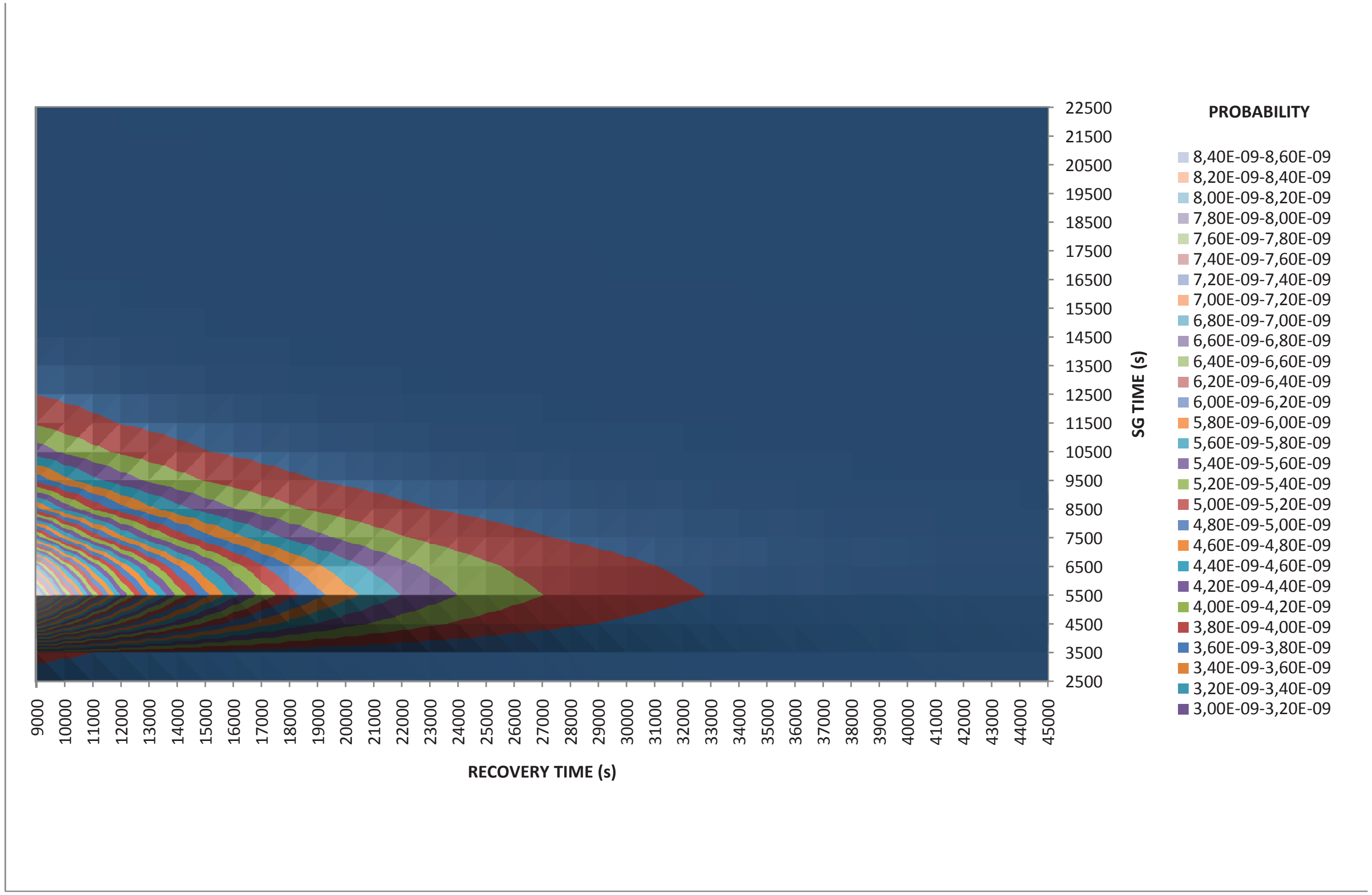


Where DD_{U0} has been found previously:



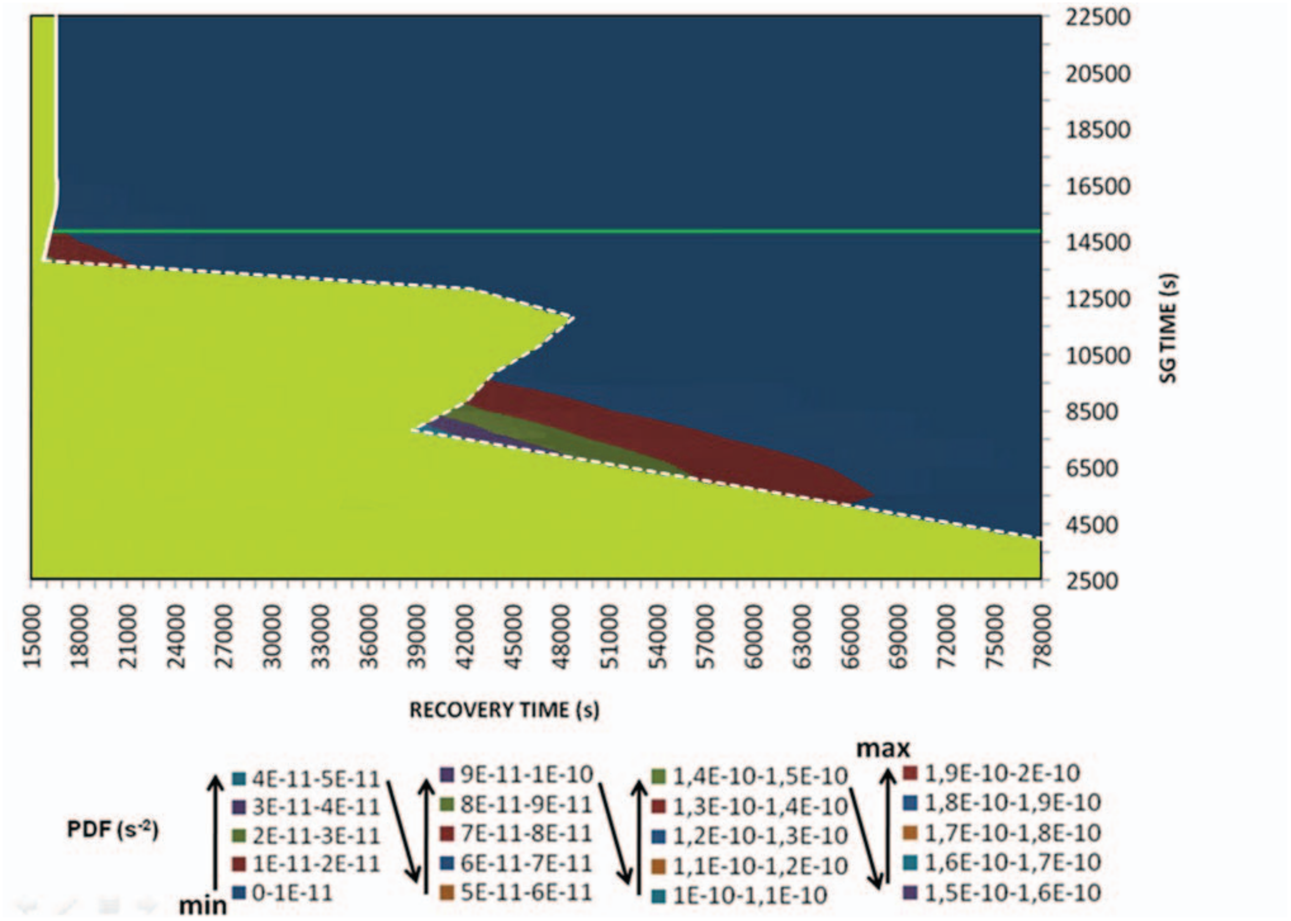
2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Step 1: 2NDH

Joint distribution of time delays,



2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Step 1: 2NDH

Joint distribution of time delays inside the Damage Domain of U0 sequence,



We can observe which region of the time-domain is more important for the probability of damage.

2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Step 1: 2NDH

The same analysis is performed for each sequence:

- Sequence U0: $H_R S A L_R$

$$\phi_{\text{dam}} = \underbrace{\phi_{\text{ini}} \cdot P_{SLOCA} \cdot (1 - P_H) \cdot (1 - P_L) \cdot (1 - P_A)}_{\text{U0 Sequence Frequency}} \cdot \underbrace{\sum_{p=1}^{DD_{U0}} \left(h_p^S \cdot h_p^{\mathsf{R}} \Delta\tau_S \Delta\tau_R \right)}_{\text{Damage Probability of U0 sequence}}$$

- Sequence U1: $H_R S a L_R$

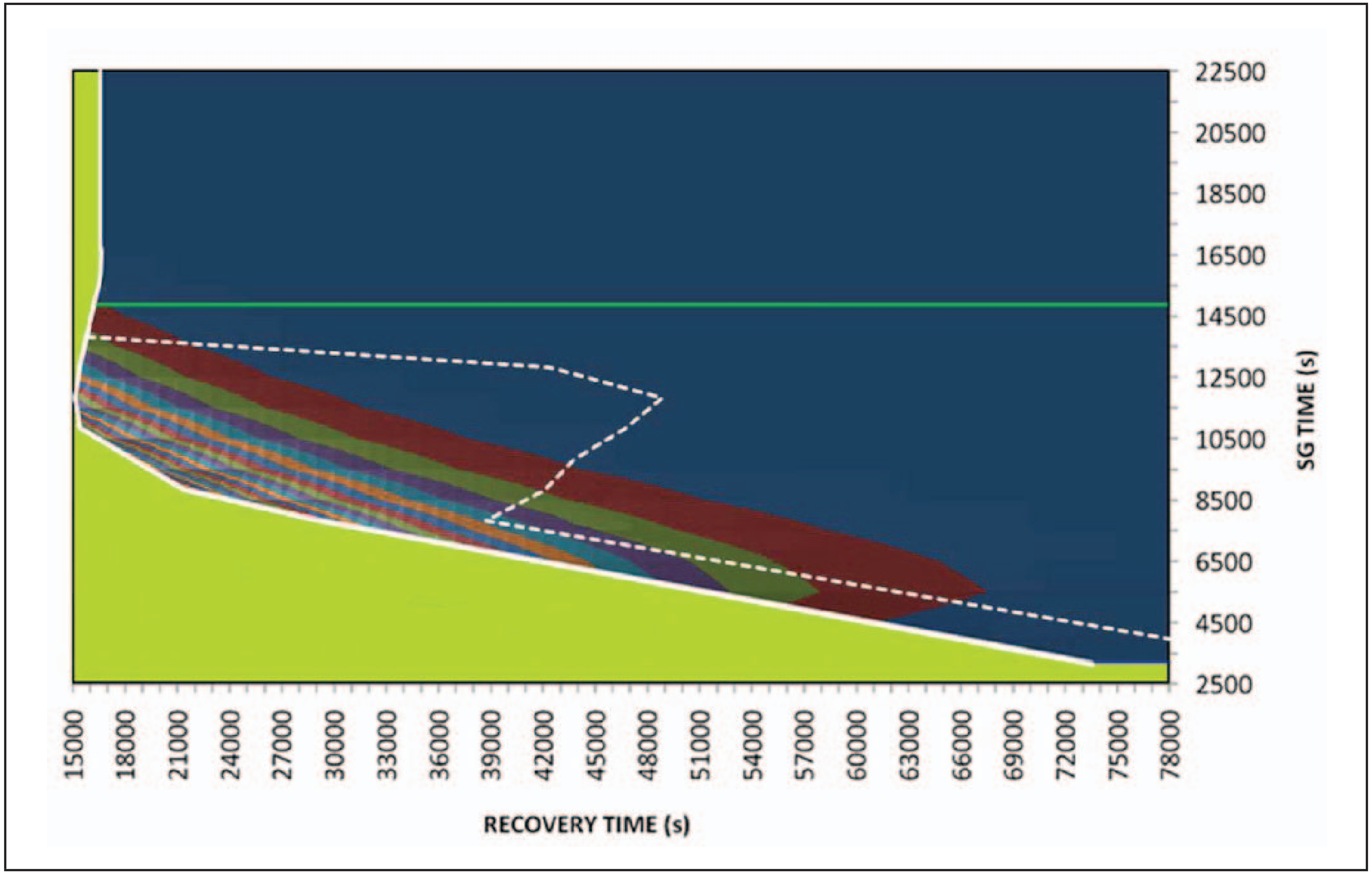
$$\phi_{\text{dam}} = \underbrace{\phi_{\text{ini}} \cdot P_{SLOCA} \cdot (1 - P_H) \cdot (1 - P_L) \cdot P_A}_{\text{U1 Sequence Frequency}} \cdot \underbrace{\sum_{p=1}^{DD_{U1}} \left(h_p^S \cdot h_p^{\mathsf{R}} \Delta\tau_S \Delta\tau_R \right)}_{\text{Damage Probability of U1 sequence}}$$

- Sequence U01: $H_R S L_R$

$$\phi_{\text{dam}} = \underbrace{\phi_{\text{ini}} \cdot P_{SLOCA} \cdot (1 - P_H) \cdot (1 - P_L) \cdot \sum_{p=1}^{DD_{U01}} \left(h_p^S \cdot h_p^{\mathsf{R}} \Delta\tau_S \Delta\tau_R \right)}_{\text{U01 Sequence Frequency= U01 Damage Frequency}}$$

2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Step 1: 2NDH

Joint distribution of time delays inside the Damage Domain of U0, U1 and U01 sequences,

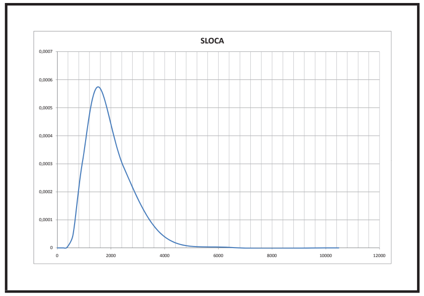


2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Step 2: 3NDH

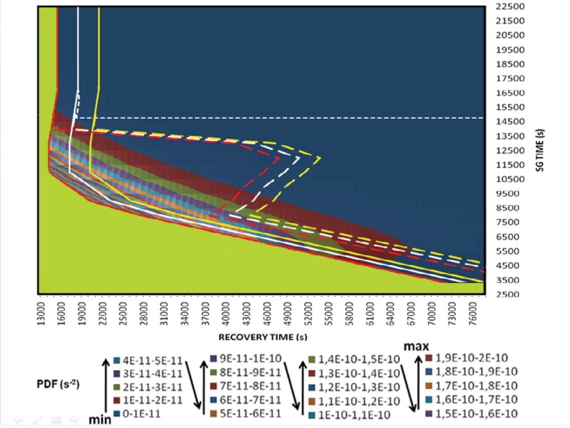
The density probability distribution of the instant of the occurrence of SLOCA is h^{SLOCA} and taking into account this uncertainty we obtain (sequence U1):

$$\phi_{\text{dam}} = \phi_{\text{ini}}(1 - P_H) \cdot (1 - P_L) \cdot P_A \cdot \sum_{z=1}^{Z} h_z^{\text{SLOCA}} \sum_{p=1}^{DD_{U0,z}} h_{p,z}^{S} \cdot h_p^{\text{R}} \Delta\tau_S \Delta\tau_R \Delta\tau_{\text{SLOCA}}$$

Where h^{SLOCA} is:



The h^{SLOCA} distribution is considered in the DD:



2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Step 2: 3NDH

There is a different DD for each value of τ_{SLOCA}. Therefore the DD is 3D.

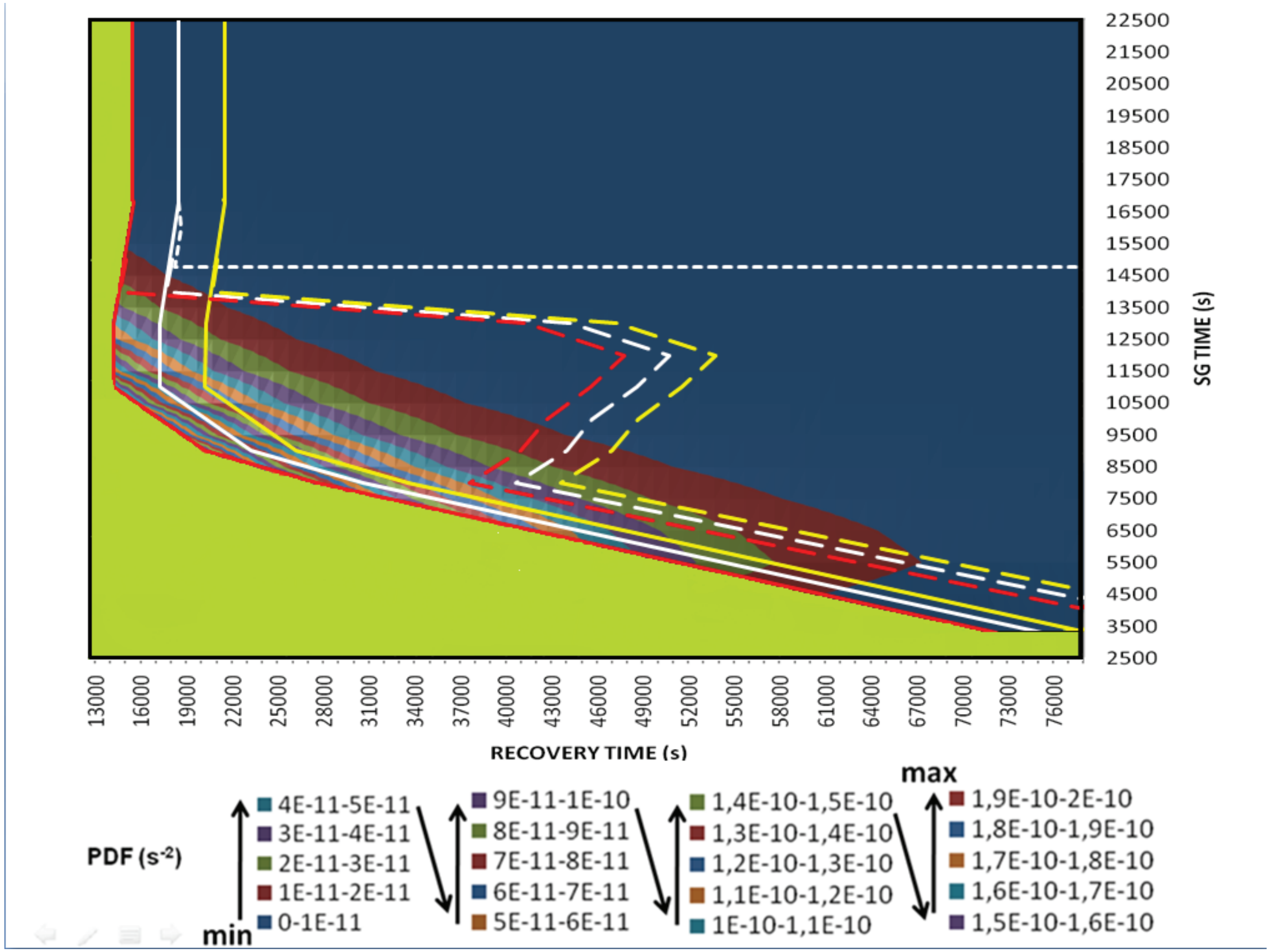


Figure 12: Damage Domain for different values of τ_{SLOCA}. TRACE Code.

2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA.
Step 3: 10% Power Uprate

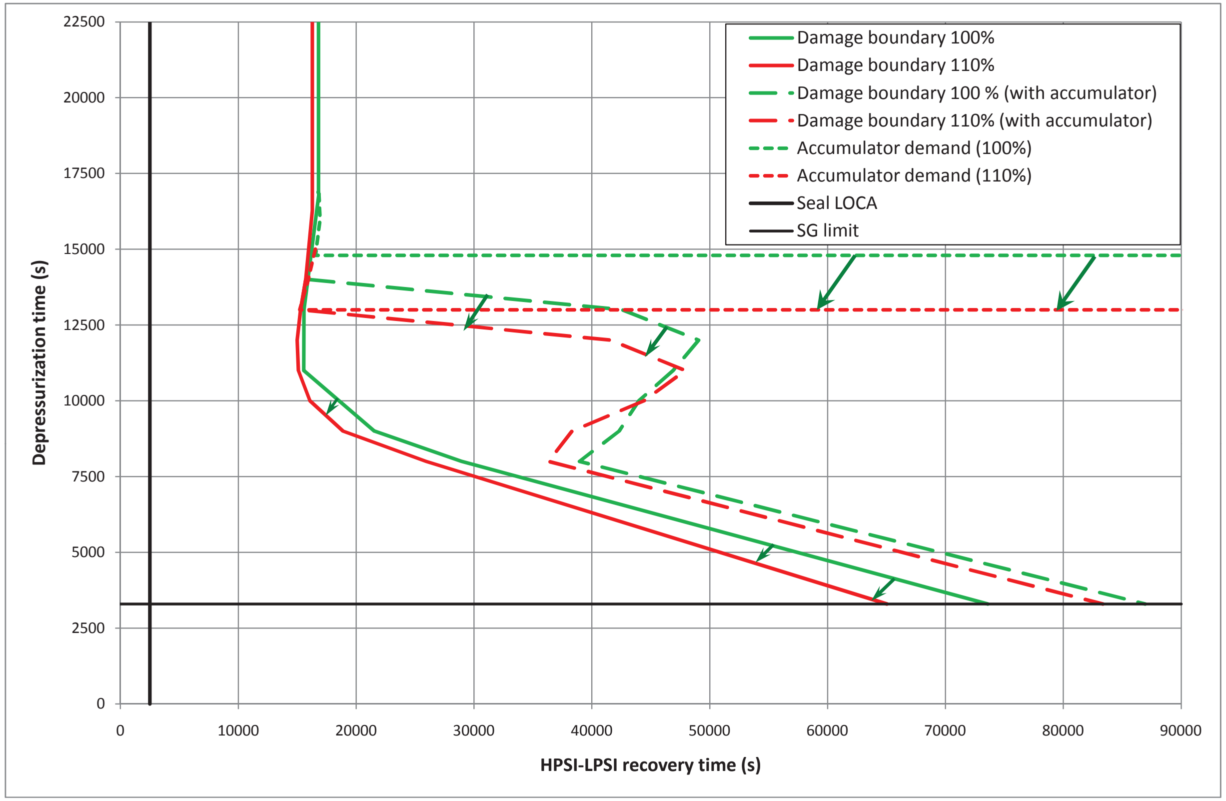


Figure 13: Increase of Damage Domain. 10% Power Uprate. TRACE Code.

2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Final Results

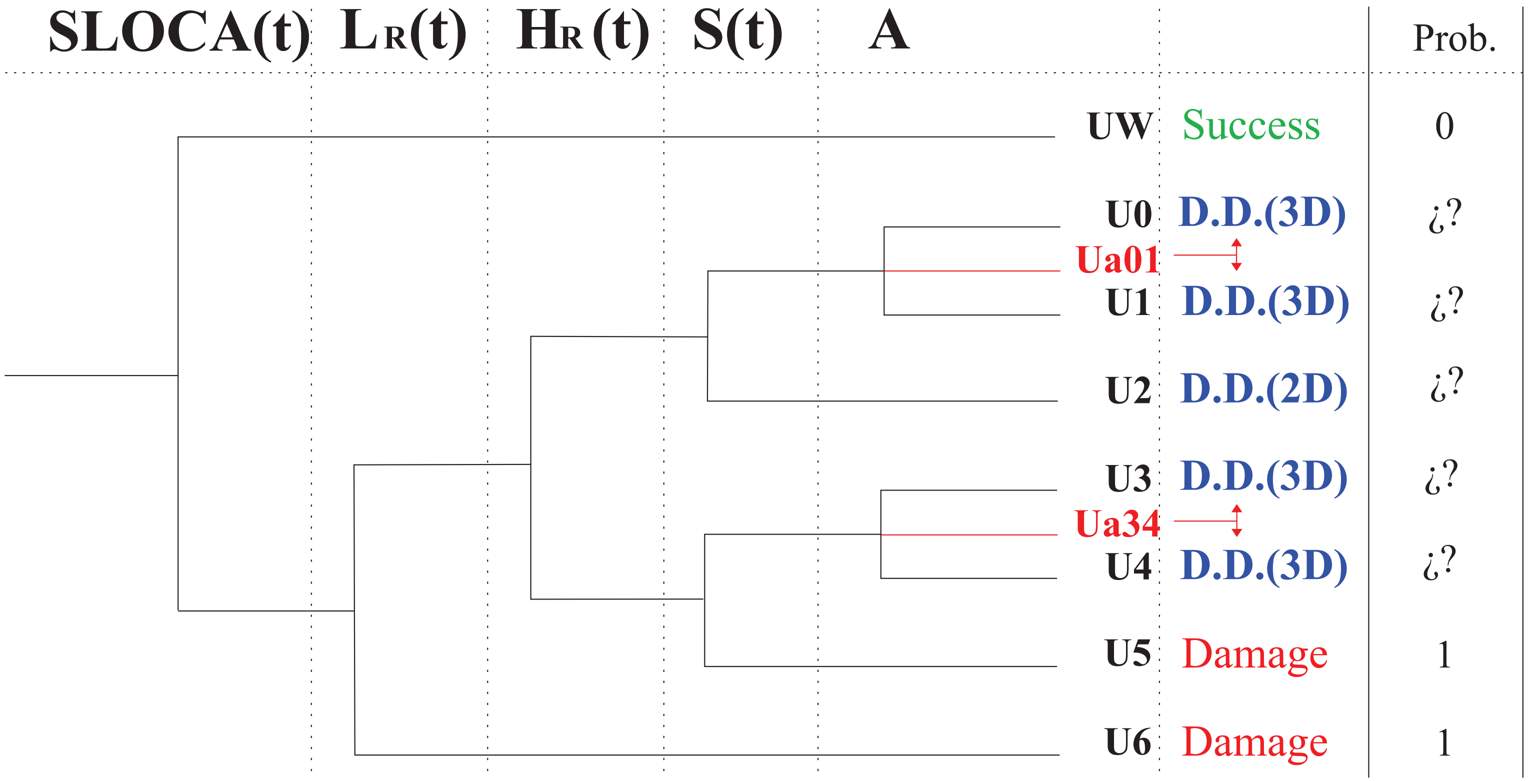


Figure 14: Generic Event Tree with Uncertainty. Three Non-Deterministic Headers

2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Final Results

Sequence of GETU	Sequence Frequency (1/year) 100%/110%	DEF(1/year) 100%/110%	Conditional exc. prob. 100%/110%
UW (R w/o SLOCA)	1.58e-03	0	0
U0 ($H_R S A L_R$)	3.94e-04	**7.83e-07/10.03e-07**	**0.0020/0.0026**
U0/U1 ($H_R S L_R$)	0.89e-07 /0.93e-7	0.93e-07 /0.97e-07	1
U1 ($H_R S a L_R$)	3.71e-07	0.04e-07 /0.06e-07	0.0108/0.0162
U2 ($H_R s L_R$)	0	0	N/A
U3 ($h S A L_R$)	9.24e-9 $<$ 1.0e-7	–	–
U4 ($h S a L_R$)	9.24e-12 $<$ 1.0e-7	–	–
U5 (hs)	0	0	N/A
U6 (l)	0.22e-7	0.22e-7	1
Total	2.0e-3	9.02e-7 /11.82e-7	0.00045/0.00059
Δ DEF		**2.80e-7 (31%)**	

Table 2: Final results for exceedance damage frequency (Results from TRACE simulations). 3 NDH.

- The results could also be obtained from MC sampling but the number of simulations is quite large because Conditional Exceedence Probability is only (0.0020, 0.0026) in U0 sequence (this point has been confirmed in several pilot applications).
- It must be remarked that we want to quantify the value of the frequency (probability) of damage, not to confirm that is lower than a given value (Order Statistics, like Wilks Approach).

2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Final Results

Here it is possible to observe where is the most important time-region for the increase of the exceedence frequency of damage.

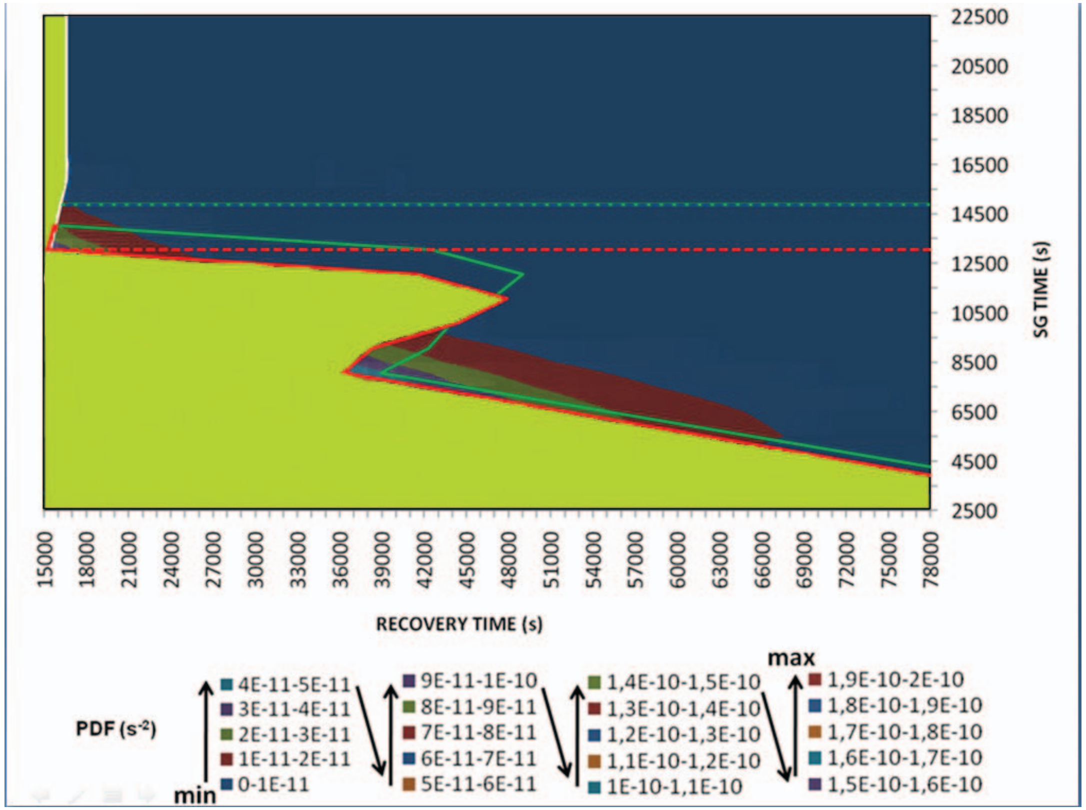


Figure 15: Increase of damage domain from 100% to 110%. U0 Sequence (HSAL). TRACE code.

2-D. RISK ASSESSMENT. Loss of CCW/SW with SLOCA. Comparison between TRACE and MAAP (3 NDH)

- Comparison TRACE / MAAP,

Sequence	Damage frequency (y^{-1}) (100%)	Damage frequency (y^{-1}) (110%)	Damage frequency (y^{-1}) increase
TOTAL MAAP	$19.63 \cdot 10^{-7}$	$20.55 \cdot 10^{-7}$	$0.92 \cdot 10^{-7}$ (5%)
TOTAL TRACE	$9.02 \cdot 10^{-7}$	$11.82 \cdot 10^{-7}$	$2.8 \cdot 10^{-7}$ (30%)

Table 3: Final results for exceedance damage frequency at 100% and 110%

- The objective of the previous analysis with MAAP is performed only with an exploratory purpose in order to obtain a *prior* of the damage domain.
- The border of the damage domain is refined with TRACE simulations.
- The results suggests that the results from MAAP code tend to be more conservative than those from TRACE code.
- It is interesting to observe that the conclusions of the application of RG 1.174 could be different with MAAP and TRACE codes because $\Delta CDF_{\text{TRACE}} > \Delta CDF_{\text{MAAP}}$

2-E. ISA Methodology Applied to SM2A Exercise. Conclusions

This ISA application to SM2A exercise has showed:

- **Its capability of analysis of DEF of the whole event tree and the decrease of Safety Margins resulting from a power uprate (Objective of SM2A exercise)**.
- This point out that it is a useful methodology in order to verify the analysis of proposed plant changes in the framework of Risk Informing Decisionmaking:
  - RG 1.174: An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.
  - RG 1.177 An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications.
- ISA methodology is very useful for the sequences in which the main source of uncertainty is the time uncertainty of human actions and/or stochastic phenomena.

  In the actual event trees there are only a few headers or phenomena with time uncertainty in each event tree.

2-E. ISA Methodology Applied to SM2A Exercise. Conclusions

- but this analysis also has showed that,
  - ISA methodology does not need to obtain the pdf of PCT, only the border of the damage domain and integrate inside it.
  - In PSA each sequence has only two possible final states, success or damage.
    However, this analysis has pointed out that it is possible to have in the same sequence a damage probability (PD) and a success probability (PS) at the same time (PD+PS=1).
  - There is a non null probability of sequences with non demanded accumulators. **This is a kind of sequence bifurcation, there are others.**
  - Available times are function of other delay times and systems availability.
- ISA methodology also allows to
  - Analyze the impact of uncertainties with probability density functions not known a priori, like failure rates which depend on the process variables.
  - Check adequacy of emergency operating procedures (EOPs) and also Severe Accident Management Guidelines (SAMG).
- **ISA Methodology is complementary to MC parameter sampling.**

3. OTHER APPLICATIONS OF ISA METHODOLOGY

ISA methodology has been applied in several projects that have been performed by UPM in collaboration with Consejo de Seguridad Nuclear:

Project	Funded by	Sequences	Plant	Code	Safety Variable	UV	Application
EC-SARNET 2004-08	EC	Hydrogen explosions	Simplified containment	Fortran models	Ignition condition Pressure containment	3	DD of pressure containment DEF
STIM 2006-10	MICINN	Cold Leg MBLOCA	PWR-W (3L)	MAAP TRACE	PCT	2	A.time DEF
CAMP OECD/ROSA OECD/ROSA-2 2007-12	CSN UNESA	CL-MBLOCA SL-MBLOCA UH-SBLOCA	Almaraz NPP PWR-W (3L)	TRACE	PCT	2	EOP adeq. A.time
SM2A 2008-11	CSN	Loss of CCW/SW	Zion NPP PWR-W (4L)	MAAP TRACE	PCT	3	Δ DEF A.time
CAMP OECD/ROSA-2 2007-12	CSN UNESA	SGTR LH-SBLOCA	Almaraz NPP PWR-W (3L)	MAAP TRACE RADTRAD	Dose PCT	2-4	DEF EOP A.Time

Table 4: Other applications of ISA Methodology

4. Next research lines.

Our next research lines with respect ISA methodology are:

- New kind of sequences: Steam Generator Tube Rupture, TLFW and SBO.
- Different success criteria for different sequences of an event tree (an application to full spectrum LOCA will be finished in a few weeks. Simulations performed with SCAIS simulation package coupled with MAAP code).
- Include new safety limits, like dose limits in SGTR sequences (at present, a preliminary DET has been obtained and it is under review).
- Parameters uncertainty sampling (Monte Carlo, stratified MC or LHS) and metamodels (?).

  New concept: There is not a Damage Domain, there is a probability of damage for each point of the time domain.
- The confidence could be included with Clopper-Pearson intervals.
- To connect SCAIS software package to the fault tree of actual PSA in order to calculate the failure probability of the headers in a dynamic way.

ACKNOWLEDGMENTS

- ACKNOWLEDGMENTS:
  These works have been funded by Consejo de Seguridad Nuclear.

Thank you for your attention!

Uncertainty and Sensitivity Analyses for CFD Codes: an Attempt of a State of the Art on the Basis of the CEA Experience

1. Uncertainties in CFD: generalities
   - Sources of uncertainties
   - Difficulties specific to CFD codes
2. The studies performed at CEA (2 studies)
3. Conclusion

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1. Uncertainties in CFD: generalities

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Sources of uncertainties in CFD

- Theoretically, the sames as for system codes, but really more numerous and miscellaneous.

- « Numerical errors »: uncertainties related to the discretization of the exact equations:
  - Time discretization errors;
  - Spatial discretization errors;
  - Iteration errors;
  - Round-off errors…

Richardson's method estimates the numerical error by:

- Using 2 or 3 more or less refined meshings;
- Knowing the order of the different numerical schemes.

But some conditions must be fulfilled:

- Polynomial behaviour of the output with respect to the size of the cells;
- No bifurcations when using different schemes.

1. Uncertainties in CFD: generalities

- Simplification of the geometry:
  - Non-controlled errors;
  - Voluntary simplifications;
  - Use of porous medium for some volumes.
- "Modeling errors": uncertainties related to the physical models:
  - Uncertainty of the constants of a model:
    Example: C_1, C_2, C_μ, Pr_k and Pr_ε of the k-ε model
  - Choice among different models:
    Example: Turbulence: RANS (k-ε or k-ω), LES, etc.
- Initial and boundary conditions:
  - Necessary hypotheses on the inlet profiles;
  - Definition of a turbulence rate at the inlet.
- User's effect:
  - For the uncertainties quoted above;
  - Analysis of the results: post-processing, considering a non-converged solution.
- Software errors: bugs, documentation. Considered as negligible.

1. Uncertainties in CFD: generalities

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Difficulties specific to CFD codes

- High CPU cost of a code run
- Large variety of the modeled objects
- Large variety of the sources of uncertainties:

A new type of input variables: the categorical variables

Example: the time scheme can be:
Explicit Euler, Runge-Kutta order 3, Cranck-Nicholson, etc.

⇨ Different choices, neither numerical, nor continuous
⇔ different levels of the categorical variable “time scheme”.

The classical measures of sensitivity:

- regression coefficients $\frac{\partial Y}{\partial X_i}$
- or correlation coefficients $\rho(X_i, Y)$

can no more be used.

1. Uncertainties in CFD: generalities

- Quantification of the uncertainty of the input variables:

The modeling errors

- Constants of the physical models:
  - Expert judgement more difficult to be applied than in the case of system codes;
  - Difficulties to find experiments suitable for inverse methods:
    - Simple experiments have too specific configurations: plane channel, homogeneous isotropic turbulence, etc.
    - Other experiments are already too complex: plumes, jets, flows with obstacle, etc.
- Choice of the model:
  - It is a categorical variable. A probability must be given to each level. Very arbitrary. Making an hypothesis of equi-probability?

The numerical errors

- Categorical variables ⇒ same problem as for the choice among different models for the physical models.

1. Uncertainties in CFD: generalities

- Methods not based on propagation of uncertainties of input parameters:

Extrapolation to the reactor case of the accuracy (the code-experiment difference) of different experiments:

  - Describing the same transient (ex: PTS)
  - At different scales.

- Main advantage:
  - Avoid the difficulties about the list of the input parameters and the quantification of their uncertainty.

- Drawbacks:
  - Need to have at one’s disposal a lot of experiments.
  - How to extrapolate the accuracy?
  - How to take into account the experimental uncertainty of the outputs?
  - Sensitivity analysis is not included in the method.

2. The studies performed at CEA

CEA approach based on propagation of uncertainty of input parameters.

- Two studies, devoted to the treatment of the categorical variables.

- Considered cases:
  - Monophasic, 2-D, laminar or turbulent;
  - CPU cost: less than 2 hours per code run.

- Input uncertain parameters:
  - Their list is plausible but not exhaustive.
  - The quantification of their uncertainty is somehow arbitrary:
  - Examples:
    - ±10% for the real variables;
    - Equiprobability for the different levels of the categorical variables.

- No comparison with experimental data for the uncertainty bands of the outputs.

- Used CFD code: Trio_U.

2. The studies performed at CEA: first study

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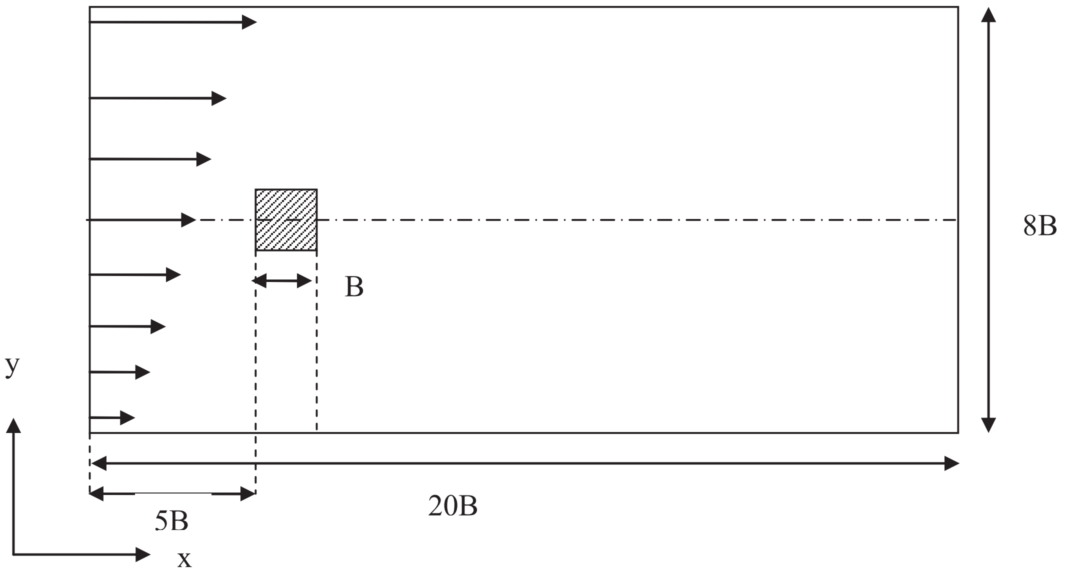
Deterministic approach based on analysis of variance (ANOVA)

- Goal: Perform sensitivity analysis despite the presence of categorical variables.
- Studied case: 2-D laminar non-stationary flow with obstacle .



2. The studies performed at CEA: first study

- 3 categorical variables:

Description of the variable	Number of levels	Numbering of the levels	Description of the level
Time scheme	4	1	Explicit Euler
		2	Runge-Kutta order 3
		3	Cranck-Nicholson
		4	Implicit diffusion
Convection scheme	3	1	Quick
		2	Centered order 4
		3	Centered order 2
Pressure solver	4	1	PCG (Preconditioned Conjugated Gradient) threshold 10^{-6}
		2	PCG threshold 10^{-5}
		3	PCG threshold 10^{-8}
		4	Cholesky

- Meshing: all the cells are identical. Their size is varied to estimate the sensitivity to the meshing.
- Other variables: of real type. Examples:

  Inlet velocity, viscosity, profile of inlet velocity, etc.

2. The studies performed at CEA: first study

- Classical tools of **AN**alysis **O**f **VA**riance (ANOVA) know how to deal with categorical variables:
  - A meta-model between the output Y and the input X_i is chosen: second order polynomial with:
    - Quadratic effects for the real variables;
    - All the interactions of order 1.
  - The Design Of Experiment (DOE) corresponding wit the meta-model is generated: D-optimal DOE with 125 points.

- Real variables:
  - A great majority of the points at the bounds of their interval of variation.
  - Few points are at the center of the interval of variation.

⇨ Deterministic method

- Categorical variables:
  - Hypothesis of equiprobability.

- The regression is built with the 125 points of the DOE.
- Results of ANOVA: Contribution to the total variance of the output Y of each effect (linear, quadratic, interaction of order 1):
  - Total of the numerical effects ≈ 25% of the total variance,
  - Viscosity = 3.7% of the total variance.

2. The studies performed at CEA: first study

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Way of doing for categorical variables:

- For each categorical variable X with n levels, (n-1) independent regression coefficients $x_1, x_2, \ldots x_{n-1}$ are calculated, by least squares:
  - When X is at its level 1, the effect of X is equal to x_1:

    x_1 must be added to the terms of the regression coming from the other input variables.
  - When X is at its level 2, its effect is x_2...
  - When X is at its level n, its effect is $x_n = -x_1 - x_2 - \cdots x_{n-1}$.

- If the DOE is orthogonal, the contribution of X to the variance of the output is:

$$\boxed{\sum_{k=1}^{n} p_k \, x_k^2}$$

p_k being the empirical frequency of the kth level (close to 1/n if equiprobability).

- Drawback: The number of regression coefficients to be calculated is increased.

2. The studies performed at CEA: second study

Probabilistic method close to that used for system codes

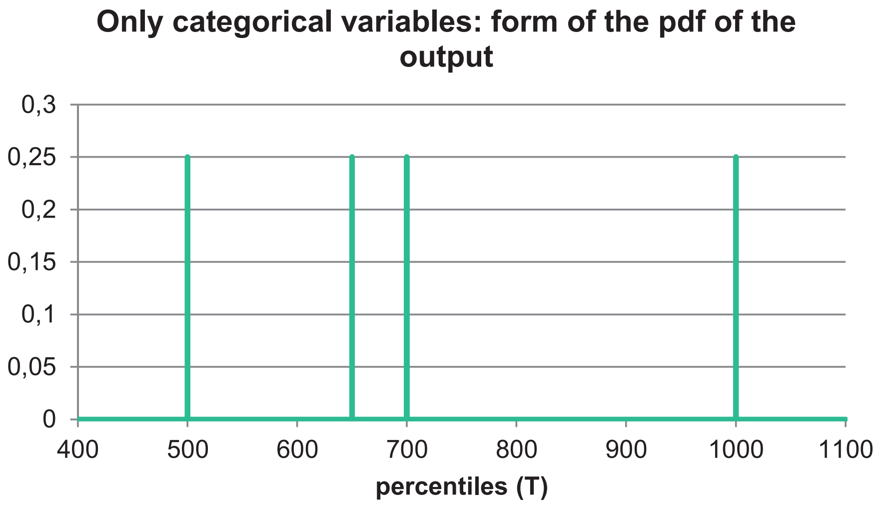
- Goal: Perform uncertainty analysis with order statistics, and simple sensitivity analysis, despite the presence of categorical variables.
- Studied case: Stationary calculation, with turbulence described via a k-ε model (not described for confidentiality reasons).

- 24 real input parameters: boundary conditions, physical properties of the fluid, constants in the k-ε model, etc.
- 3 categorical variables:
  - 3 different configurations for the initial conditions;
  - 3 levels for the time scheme;
  - 2 types of outlet conditions.

2. The studies performed at CEA: second study

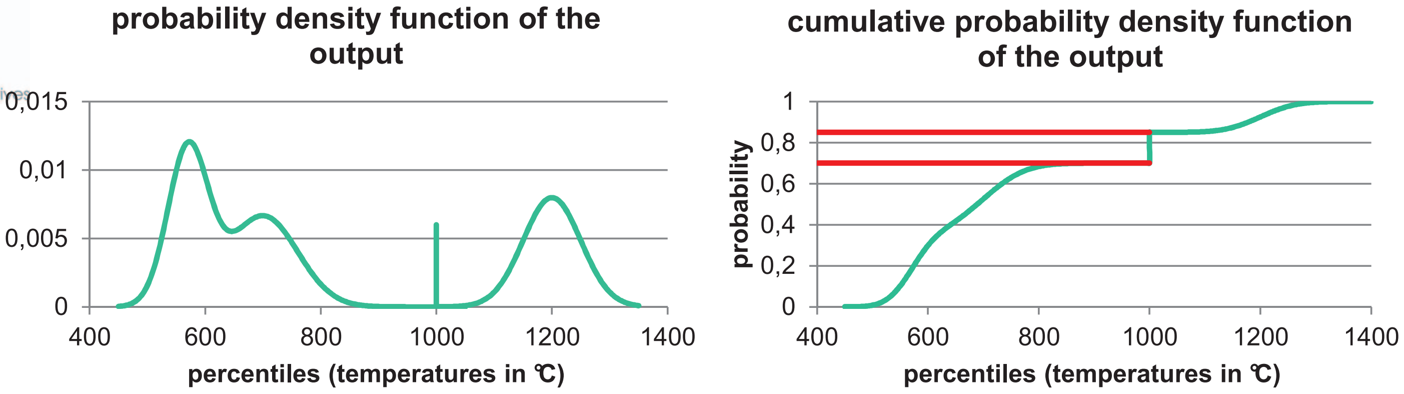
Uncertainty analysis:

- 100 Trio_U code runs are performed by sampling the values of the input parameters with a Simple Random Sampling (SRS) Design of Experiment (DOE).
- Order statistics are used to obtain percentiles (ex: 5% and 95% percentiles).
- The only hypothesis to use order statistics: the probability density function (pdf) of the output must be continuous. Not checked if there are only categorical variables:



2. The studies performed at CEA: second study

- Example of theoretical results with a mixing of real and categorical variables:



probability density function of the output

cumulative probability density function of the output

- For a given level of a categorical variable, the continuous real variables have no effect.

⇨ Each time this level is sampled, all the values of the output are equal: there is peak in the pdf of the output.

- Considered case: all the 100 values of the output are different, the pdf of the output is perfectly continuous.

2. The studies performed at CEA: second study

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Sensitivity analysis: 2 methods:

- Method 1 based on regression coefficients: The same as that used in the first study, but by considering a linear regression.

Other difference: SRS sampling, taking into account the pdf of the inputs ⇒ probabilistic method.

- Method 2 based on correlation coefficients:
- For the real variables X:

Contribution of X to the total variance of Y = $\boxed{\rho^2 \times v\hat{a}r(Y)}$

- For the categorical variables X:
  - p_k = empirical frequency of the kth level (close to 1/n if equiprobability).
  - e_k = mean value of Y when X is at its level k.
  - $\bar{e}$ = mean value of the n e_k.

Contribution of X to the total variance of Y = $\boxed{\sum_{k=1}^{n} p_k (e_k - \bar{e})^2}$, i.e.:
the conditional variance of the expectation value of Y knowing X: $var[E(y/X)]$

- Both methods give the same dominant parameters.

Measures of sensitivities are very close for the outputs of type scalar at a given time and at a given point.

3. Conclusion

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PART 1: Generalities about uncertainties in CFD

With respect to the system codes:

- More sources of uncertainties;
- More systematical presence of categorical variables:

  Time schemes, convection schemes, type of boundary conditions, etc.
- More difficulties to quantify the uncertainties of the input variables, especially the levels of probability of the categorical variables;
- Those difficulties can be avoided by extrapolating the accuracy of the outputs, obtained in different experiments devoted to the same kind of transients. But:
  - Are always experiments available?
  - How to extrapolate the accuracy?
- Higher CPU cost.

3. Conclusion

PART 2: Studies performed at CEA

- CEA method based on propagation of the uncertainty of input variables.
- Two studies tackling the difficulty of the presence of categorical variables.
- Difficulties specific to the categorical variables:
  - Prevent to use classical sensitivity measures;
  - Introduce a certain discontinuity in the pdf of the output.

Forthcoming studies at CEA

1. Using the second (probabilistic) approach for a case with experimental data.

- Goal: check that the uncertainty bands envelop the experimental data.

  ⇒ A special attention paid to the choice of the input parameters.

2. Solving the problem of the high CPU cost of a code run:

- Auto-regressive model between two kinds of modelings:
  - A “cheap” and an “expensive” modeling, corresponding for example with a coarse and a refined meshing.
  - Kriging for each modeling.

Question still pending: Quantification of the uncertainty of the input parameters, especially if they are of categorical type.

GOBIERNO DE ESPAÑA
MINISTERIO DE CIENCIA E INNOVACIÓN
Ciemat
Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas

OECD/CSNI Workshop on Best Estimate Methods and Uncertainty Evaluations

Preliminary insights of the impact of steady state uncertainties on fuel rod response to an RIA

Anne Huguet & Luis E. Herranz(*)
presented by Claudia López
(*) *luisen.herranz@ciemat.es*

Scope of the study

- Exploration of UA methods applied to Nuclear Fuel Behavior studies
  - Powerful analytical tool
  - Adding value and skill
- Application to postulated RIA scenario
  - Experience on RIA modeling within the team
  - Scarce literature (emphasis on LOCA)

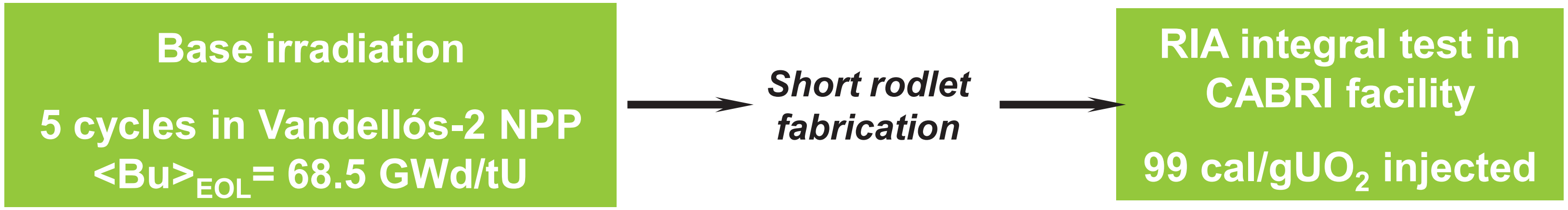
UA Methodology

1. Problem identification
2. Uncertainty sources
3. Importance assessment
4. Uncertainty propagation
5. Sensitivity analysis

1. Problem specification

Scenario

- Case study = CIP0-1 test
  - What?



Base irradiation

5 cycles in Vandellós-2 NPP
$\langle Bu \rangle_{EOL}$= 68.5 GWd/tU

Short rodlet fabrication

RIA integral test in CABRI facility

99 cal/gUO$_2$ injected

  - Why?
    - Prototypical PWR RIA
    - Well characterized experimentally
    - Fully documented
    - Previous studies

1. Problem specification
Tools

- Case study = CIP0-1 test
  - How?
    - **FRAP family codes**

Base irradiation

FRAPCON-3.4a

→ ***Restart file*** →

Transient

FRAPTRAN1.4

- **Starting point = a BE reference case**
  - Both BE codes
  - Most of the input adjusted to experimental data

1. Problem specification

Focus

- Study of the impact of FRAPCON-3 input uncertainties on FRAPTRAN predictions:
  - Up to which accuracy we need to go in the steady state modeling for subsequent RIA modeling?
  - Which accuracy can we reach in the RIA modeling?

Rodlet irradiation
FRAPCON-3.4a

RIA transient
FRAPTRAN1.4

1. Problem specification

UA settings

- FRAPTRAN outputs of interest
  - Which?
    - Maximum clad elongation
    - Permanent hoop strain at PPN
    - Peak SED
  - Why?
    - Bases of clad failure prediction models

2. Uncertainty sources

Identification

- Data related to transient
  - – Scenario description
  - – Modeling approach
- Data related to initial state of the rodlet
  - – Base irradiation
    - **1. FRAPCON-3 inputs deck data**
    - **2. FRAPCON-3 steady-state models**
  - – Refabrication process

2. Uncertainty sources

Methodology

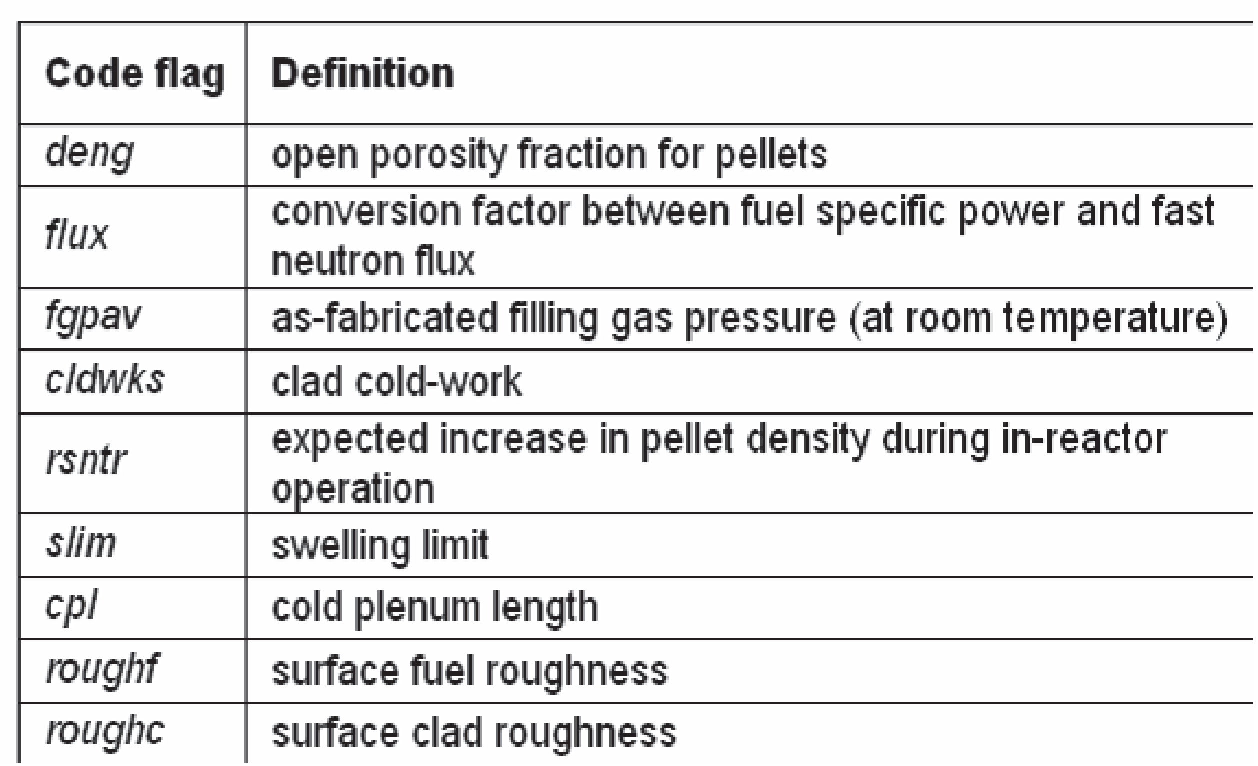
- Identification based on:
  - An extensive study of the codes transmission chain
  - An expert judgment

- Quantification:
  - Relies on data availability, expert recommendation, code user experiment
  - Choice of realistic bounding intervals for each variable

2. Uncertainty sources

Input data

9 inputs parameters:



Code flag	Definition
deng	open porosity fraction for pellets
flux	conversion factor between fuel specific power and fast neutron flux
fgpav	as-fabricated filling gas pressure (at room temperature)
cldwks	clad cold-work
rsntr	expected increase in pellet density during in-reactor operation
slim	swelling limit
cpl	cold plenum length
roughf	surface fuel roughness
roughc	surface clad roughness

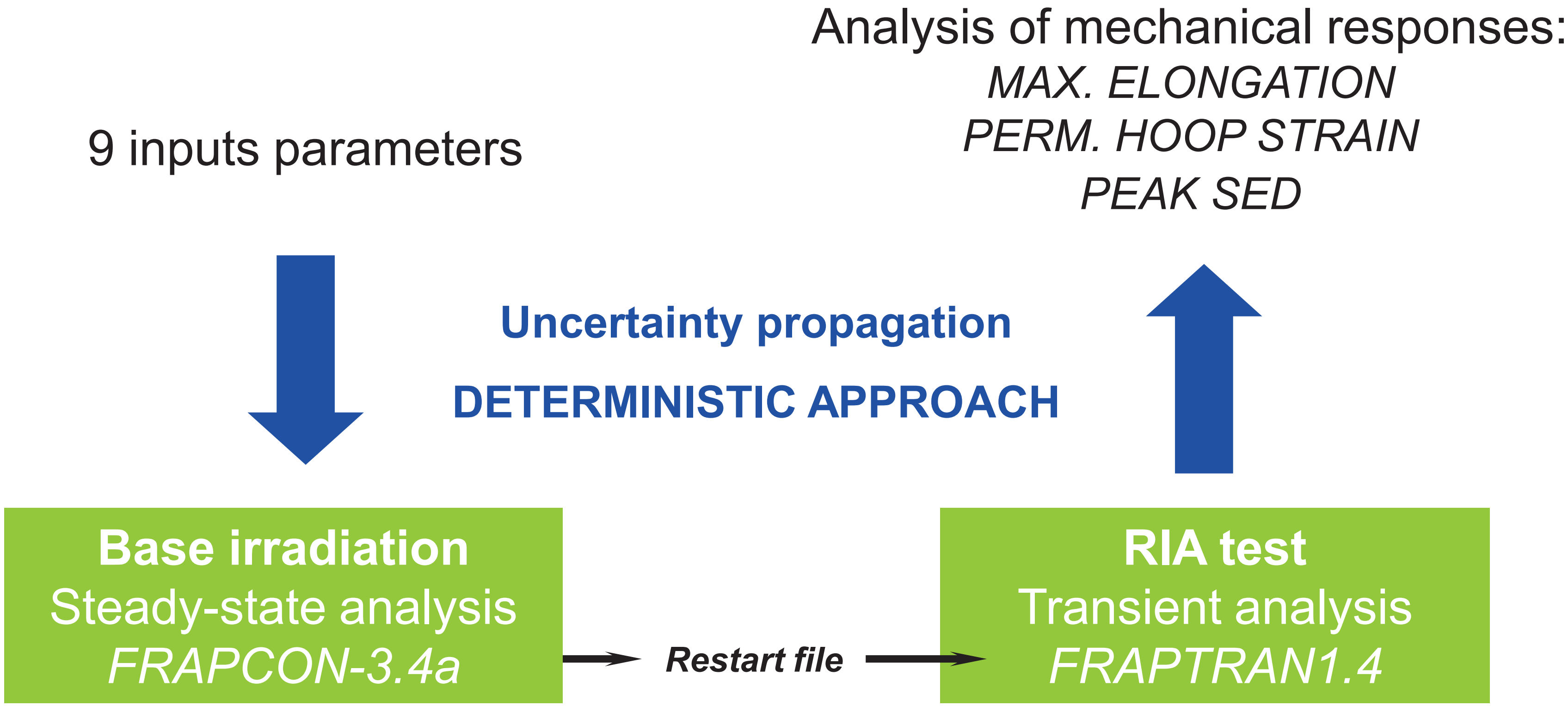
2. Uncertainty sources

Input data

Code flag	Unit	BE value	Lower bound	Upper bound	Rationale
deng	%TD	0	0	1.5	Literature
flux	n/m^2s per W/g	2.21E+16	1.11E+16	4.42E+16	Expert advise
fgpav	Pa	2.35E+06	2.28E+06	2.42E+06	Literature
cldwks	ND	0.5	0.1	0.8	Code limits
rsntr	kg/m^3	56.656	0	105.31	Developer recommendation
slim	vol. frac.	0.05	-	0.1	Expert advise
cpl	m	0.047	0.037	0.057	Literature
roughf	μm	4	0.25	14.4	Literature
roughc	μm	6	0.17	4.5	Literature

3.Uncertainty propagation

Approach



9 inputs parameters

Analysis of mechanical responses:
MAX. ELONGATION
PERM. HOOP STRAIN
PEAK SED

Uncertainty propagation

DETERMINISTIC APPROACH

Base irradiation
Steady-state analysis
FRAPCON-3.4a

Restart file

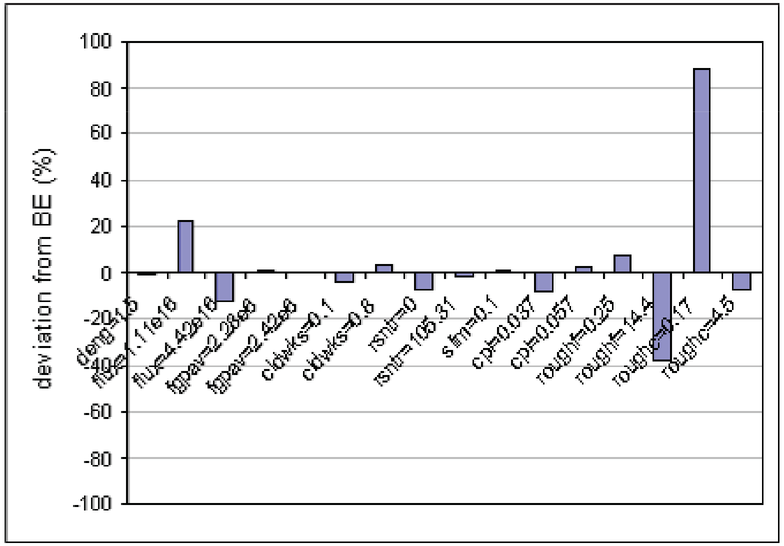
RIA test
Transient analysis
FRAPTRAN1.4

3. Uncertainty propagation

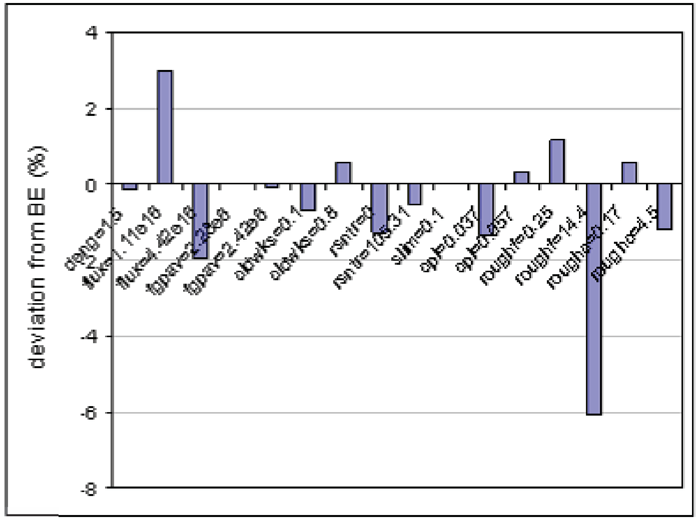
Methods

- Which?
  - **Qualitative = One-At-a-Time (OAT)**
    - Perturbation of only one factor per computer run
    - Efficient and intuitive importance assessment method
  - **Quantitative = Response Surface Method (RSM)**
    - Approximation of the original code response by an analytical "function"
    - Fitting of code responses sample
    - Meta-model qualification
    - Probabilistic uncertainty propagation on meta-model
- Why?
  - Simple (explorative study)
  - Pdf unknown

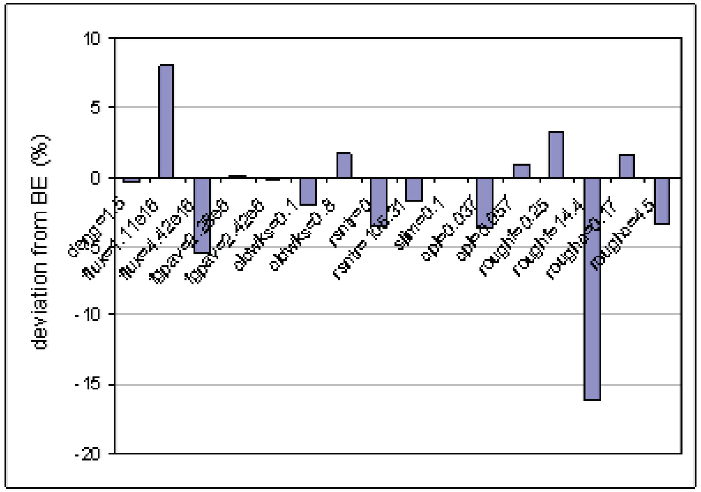
4. Importance assessment. *OAT method*



(a) Clad permanent hoop strain at PPN



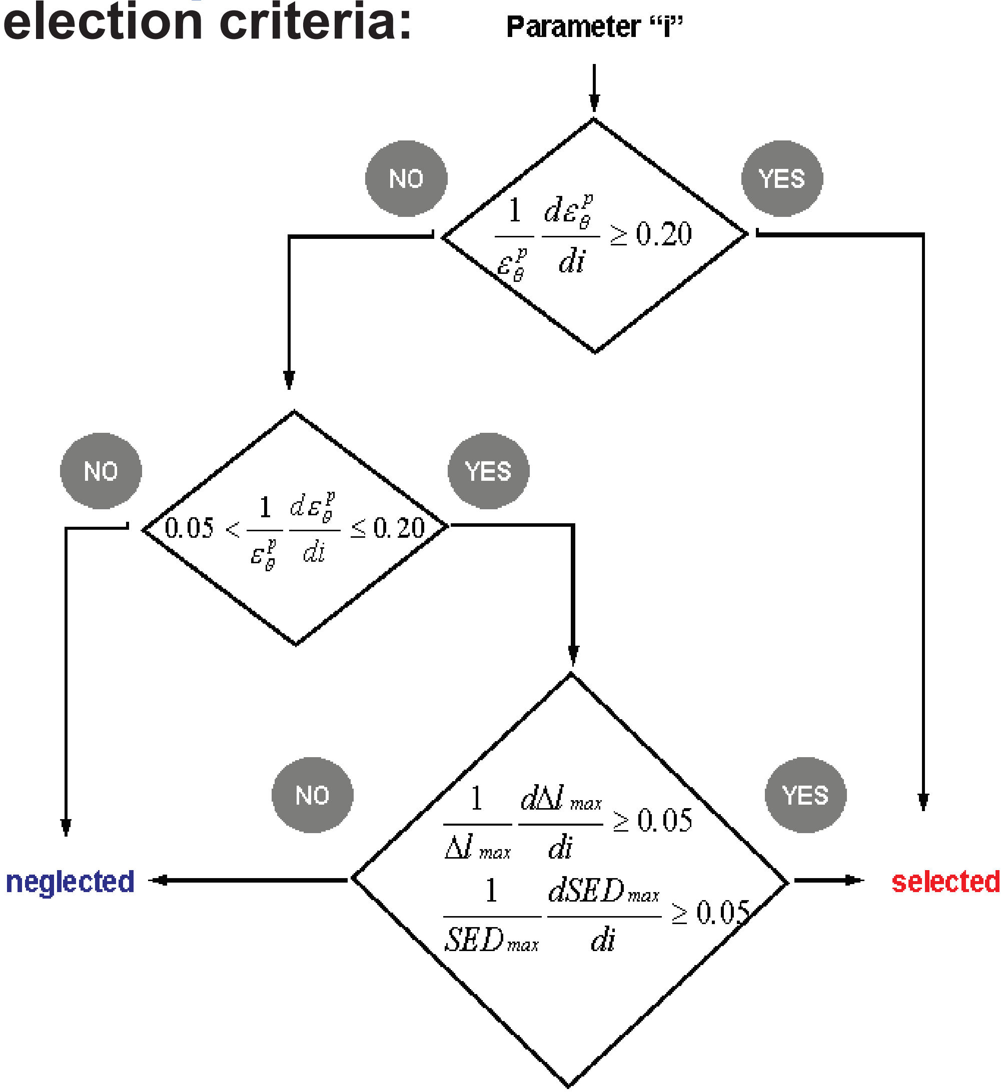
(b) Maximum clad elongation



(c) Peak PNNL-SED value at PPN

4. Importance assessment. *OAT method*

Selection criteria:

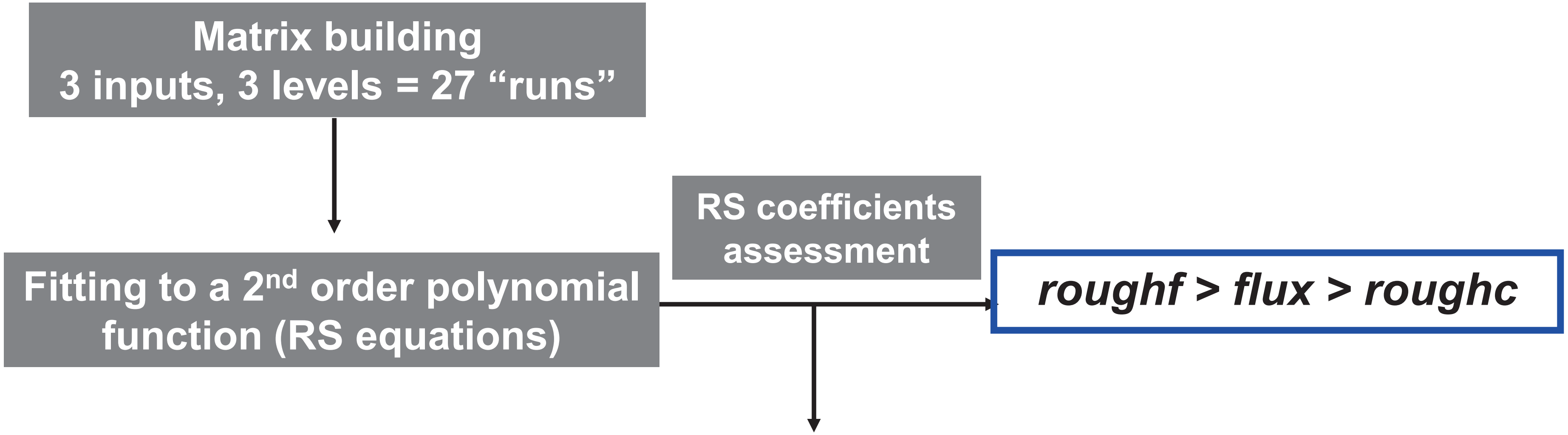


Identification of 3 main dominant parameters:

flux, roughf, roughc

5. Sensitive analysis

RSM method



Matrix building
3 inputs, 3 levels = 27 "runs"

Fitting to a 2nd order polynomial function (RS equations)

RS coefficients assessment

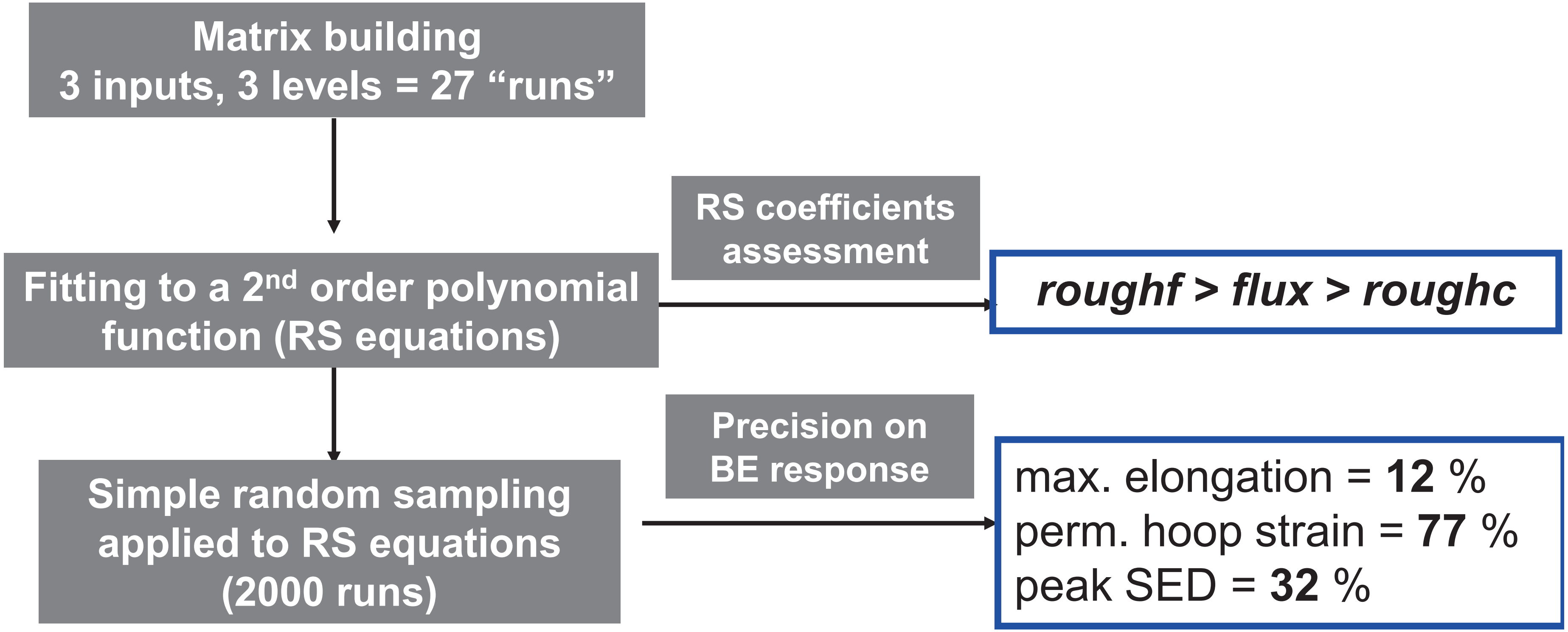
roughf > flux > roughc

Meta-model: choice of a 2nd order polynomial function:

$$z_i = z_0 + \sum_i a_i x_i + \sum_{i,j} a_i a_j x_i x_j$$

5. Sensitive analysis

RSM method



Matrix building
3 inputs, 3 levels = 27 “runs”

Fitting to a 2nd order polynomial function (RS equations)

RS coefficients assessment

roughf > flux > roughc

Simple random sampling applied to RS equations (2000 runs)

Precision on BE response

max. elongation = **12** %
perm. hoop strain = **77** %
peak SED = **32** %

Final remarks

- Set-up of a UA methodology applied to RIA modelling
- Upper and lower intervals of key variables might affect substantially the results
- Fuel and cladding roughness deserve attention when building up the input deck
- Uncertainties of steady state variables affect key variables of the transient in quite a different way
- Next steps

  - Similar analysis with FRAPCON-3 models
  - Then combining both

Acknowledgements

This work is framed within the CSN-CIEMAT agreement on “Thermo-Mechanical Behavior of the Nuclear Fuel at High Burnup”.

Thank you for your attention