NRC FOREIGN TRIP REPORT

Subject

Meetings with the OECD/NEA ad hoc Group of Experts to Develop a CSNI Action Plan in the Area of Safety Margins, November 21-22, 2002.

Dates of Travel and Countries and Organizations Visited

November 19-23, 2002
Paris, France
OECD/NEA sponsored program.

Author, Title, and Agency Affiliation

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Sensitivity

N/A

Background or Purpose

The NRC participates in several working groups organized under CSNI including the Special Expert Group on Fuel Safety Margins (SEGFSM) and the Working Group on Analysis and Management of Accidents (GAMA). The subject meeting was requested by CSNI to develop an action plan to consider safety margins in nuclear power plants.

Abstract: Summary of Pertinent Points and Issues

A two day meeting was held at the NEA Offices in Issy-les-Moulineaux, France to discuss the topic of safety margins in nuclear plants and to develop an action plan to more fully define and understand safety margins. One of the issues considered was the definition of safety margin itself, and the various sources of safety margin. The product of the meetings was an Action Plan that if enacted would produce a report on safety margin issues and how they are affected by simultaneous changes in plant operating conditions. The ad hoc group recommends to CSNI that a multidisciplinary Task Group be formed to (a) advise CSNI and CNRA on safety margin issues, (b) review safety margin relevant work performed by the CSNI working groups, and (c) propose to CSNI coordinated safety margin activities.

Discussion

The meeting followed the attached agenda, with M. Durin (IRSN) serving as Moderator for the first day's meeting. The background of the meeting and relation of the proposed work to that performed by other CSNI working groups was discussed by J. Royen (OECD). The main difference between the group and Action Plan developed in this meeting is that it is intended to be cross-cutting and multidisciplinary in scope. While safety margins are often considered in

terms of the peak cladding temperature (PCT) and the primary system thermal-hydraulic analysis, the objective of this group is to consider safety margin issues across several areas such as fuel safety, containment, human actions, etc. and synergistic relationships between these areas. Current activities performed under CSNI related to safety margins were then presented by M. Hrehor on the Special Expert Group on Fuel Safety Margins (SEGFSM) and by M. Durin on the Working Group on Analysis and Management of Accidents (GAMA). J. Royen presented brief summaries of Working Group on Risk Assessment (WGRISK) and the Working Group on Integrity and Ageing of Components and Structures (IAGE).

The part of the meeting consisted of presentations by all delegates on current activities within their respective organizations on safety margin issues (power upratings, operating cycles, fuel burnup, examination of fault sequences, etc.). Handouts are included in this trip report.

I presented a summary of several NRC research activities that are related to safety margin including the High Burnup Fuel Research Program and Risk Informed Regulation. A copy of the presentation is included in this attachment.

Following presentations by the delegates, the attendees developed the following Action Plan:

Task 1: Definition of Safety Margins and Related Concepts (duration: 6 months, covering meetings 1 and 2)

Sub-task 1A:a) define types of safety margins
b) characterize <u>sources</u> of safety margins

Sub-task 1B:a) identify parameters vs. acceptance criteria to quantify safety margins for Design Basis Accidents (DBA)
b) identify parameters vs. acceptance criteria to quantify safety margins for Beyond Design Basis Accidents (BDBA)

Sub-task 1C: Define related definitions: such as "conservative," "best estimate," "realistic," etc.

Task 2: Safety Margin Evaluation Methods (duration: 6 months, covering meetings 1 and 2)

Sub-task 2A: identify and summarize methods used to deal with safety margins
[e.g., "very conservative" (Appendix K), Best Estimates + uncertainties, "bounded Best Estimates," qualitative, PSA, etc.]

Sub-task 2B: identify all areas of consideration
(e.g., fuel design, thermal-hydraulics, containment behavior, structural behavior, human factors, neutron kinetics, etc.)

Task 3: Dynamic Aspects of Safety Margins
(duration: 2 years, covering meetings 1, 2 3 and if necessary 4)

Sub-task 3A: identify the links to PSA

Sub-task 3B: identify bounding scenarios

Sub-task 3C: identify impact of operator actions (human factors)

Sub-task 3D: identify minimum operator or system response time of requirements

Task 4: Quantification of Safety Margins (duration Sub-task 4A: 3 to 6 months, covering meetings 1 and if necessary 2)
(duration Sub-task 4B : continuing task)

Sub-task 4A: identify and list those areas that quantify safety margins
(e.g., fuel design, thermal-hydraulics, containment behavior, structural behavior, human factors, neutron kinetics, etc.)

Sub-task 4B:review safety margin relevant work performed by CSNI WGs/SEGs and other groups; identify and ask questions with CSNI WGs/SEGs (experts from these groups should attend Task Group meetings as necessary, depending on the agenda of the meetings); recommend the organization of workshops and other appropriate meetings

Task 5: Appropriate Uses of Safety Margins (duration: 2 years, covering meetings 3, 4, 5 and 6)

Sub-task 5A: identify and list examples of integrated use of safety margins

Sub-task 5B: discuss overall impact on safety of margin reductions; discuss safety margin interactions

Sub-task 5C: develop guidance for use of safety margin tools

Pending Actions and Planned Next Steps for NRC

The next meeting, assuming approval of the plan by the Technical Program Group, will be in March at the NEA Offices in Paris to initiate work on the report and activities in the Action Plan.

Attachments to Foreign Trip Report

25 October 2002

OECD NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

TENTATIVE AGENDA

MEETING TO DEVELOP A CSNI ACTION PLAN IN THE AREA OF SAFETY MARGINS

**(NEA Offices, Issy-les-Moulineaux, France
21-22 November 2002)**

I. A meeting **to develop a CSNI Action Plan in the Area of Safety Margins** will be held on **21-22 November 2002** at the NEA Offices, Le Seine Saint-Germain, 12 Boulevard des Iles, F-92130 Issy-les-Moulineaux, France, in room A on the 7th floor. It will start at **09.00 a.m.** on the first day. Upon arrival, participants will need to register with the receptionist on the 6th floor.

II. The following agenda is proposed:

1. Welcome. Opening remarks.

2. Choice of a meeting Moderator.

3. Adoption of the agenda. (Moderator)

4. Background and purpose of the meeting. Requests made by CSNI and CNRA. Need for an integrated approach (addressing safety margins associated with all aspects of nuclear power plants). (Moderator)

5. Work done in the past under CSNI sponsorship, in particular in the framework of PWG2 on Coolant System Behaviour. (M Réocreux and/or H. Glaser)

6. Current work performed under CSNI sponsorship:

   - Special Expert Group on Fuel Safety Margins (SEGFSM). (M. Hrehor)
   - Working Group on Integrity and Ageing of Components and Structures (IAGE). (E. Mathet)
   - Working Group on Risk Assessment (WGRISK). (B. Kaufer)
   - Working Group on Analysis and Management of Accidents (GAMA). (M. Durin)

7. Current and planned national activities. Presentations (all delegates) and Round table discussions on:

    - Good practices in the identification and management of safety margins for operating nuclear power plants.
    - Practical approaches to quantifying safety margins in various areas, and discussion of possible synergistic effects:
        - Power upratings.
        - Longer operating cycles.
        - Fuel design.
        - Increased fuel burnup.
        - Emergency core cooling systems performance in design basis accidents.
        - Examination of fault sequences limiting plant operation.
        - Possible impacts of plant ageing and plant life extension.
        - Other areas.
    - Appropriate use of safety margins.
    - Role of PSA.
    - Possible revisions of acceptance criteria.

8. Identification of the problems. Objectives of the Action Plan. Expected Products. (Moderator)

9. Development of a CSNI Action Plan in the area of Safety Margins:

    - Setting up of a multidisciplinary Task Group.
        - Mandate of the Group.
        - Specific actions, priorities and time scale:
            - To be undertaken by the Task Group.
            - Proposed for integration into CSNI WG/SEG programmes.
    - Future Workshops (identification of topics, time frame).

10. Other matters.

11. Next meeting (if necessary) (dates, place, main points of the agenda).

III. It would be helpful if the various contributions made under item 7 of the agenda would be supported by brief written notes.

IV. Information on accommodation, and on transportation to the NEA building, can be found on the NEA Internet site (at 'http://www.nea.fr'), in particular under the headings Helpful Hints (transportation, hotels) and Information for New Staff (various routes to the NEA). Should you need additional information, please get in touch with Jacques Royen ('jacques.royen@oecd.org') or Katia Van de Putte ('katia-karina.vandeputte@oecd.org').

U.S. NRC Research Activities on Safety Margins

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Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission

Meeting to Develop a CSNI Action Plan in the Area of Safety Margins

21-22 November, 2002

SAFETY MARGIN

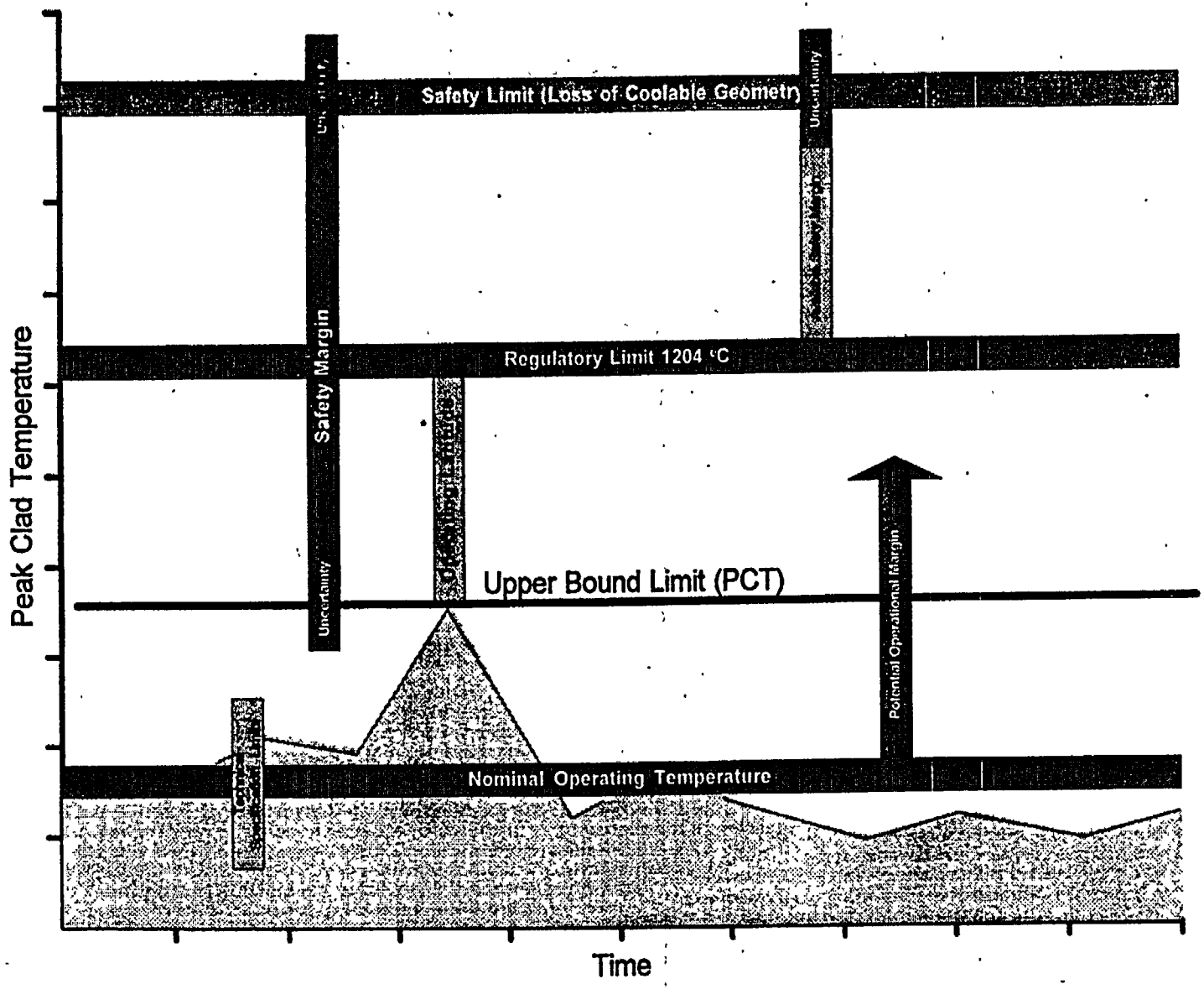
- The term "safety margin" can have different meanings, depending on point of view:

  Regulatory Agency: Margin is difference between a regulatory limit (acceptance criteria) and a safety limit (failure condition).

  Vendor / Licensee: Margin is difference between predicted performance and regulatory limit.

- Total or Overall Margin = Sum of safety margins reserved by Regulatory Agency plus margin used within analysis methods.

SAFETY MARGIN: Example



Practical Approaches & Trends to Quantifying Safety Margins

Power Upratings:

Most PWRs in the U.S. have, or plan to uprate by as much as 10%. Many BWRs will uprate power by as much as 20%.

Best Estimate analysis methods are generally used to obtain & quantify safety margin.

Practical Approaches & Trends to Quantifying Safety Margins

Increased Fuel Burnup

Major NRC/RES research effort is underway to investigate cladding integrity for high burnup fuel at Argonne National Lab.

Tests Include:

- LOCA Behavior Tests (Oxidation, Post Quench Ductility, ...)
- Dry Storage Behavior (Creep)
- Mechanical Properties following RIA

Risk Informed Regulation

- Addresses several issues where conservative assumptions result in loss of overall margin.

  - **Equipment Availability**
  - **Break Size**
  - **Off-Site Power Availability Assumptions**

STUDY OF RISK-INFORMED CHANGES TO ECCS REQUIREMENTS

Each LWR must be provided with an ECCS

ECCS Reliability	Assure system safety function can be accomplished	• onsite power operation (offsite power unavailable) and assuming a single failure; and • offsite power operation (onsite power unavailable), and assuming a single failure
ECCS Acceptance Criteria	Criteria for ECCS cooling performance following postulated LOCA	• Peak cladding temperature ≯2200°F • Maximum cladding oxidation ≯0.17 times before oxidation • Maximum H2 generation ≯0.01 of all metal reaction • Coolable core geometry • Long term cooling
ECCS Evaluation Model	ECCS cooling performance calculated with acceptable evaluation model	• Realistic (best-estimate) including assessment of uncertainties • With required and acceptable features of Appendix K
ECCS LOCA size definition	ECCS cooling performance calculated for number of LOCA sizes and locations	Accidents result in loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS.

Possible Revisions of Acceptance Criteria

- Fuel research program expected to provide data to justify "risk-informed" acceptance criteria for 10 CFR 50.46
  - **Peak Cladding Temperature**
  - **Maximum Local Cladding Oxidation**
- Appendix K Relaxation Considered
  - **1994 Decay Heat Standard**
  - **Cathcart-Pawel for Metal-Water Reaction**
  - **Eliminate Steam Cooling Requirement**
  - **Eliminate Prohibition on Return to Nucleate Boiling**

Advanced Plant Safety Margin

- New plant designs require a re-thinking of safety margins and design basis accident scenarios
- Under consideration: AP1000, ESBWR, SWR-1000, ACR-700, IRIS, MHTGR, PBMR

Three main types of BWRs:

- GE
- KWU and ABB Internal Pump reactors
- External Pump Reactors

Variety of Operating Limit and Reactor Protection Strategies in order to:

- Optimize Fuel Economy
- Uprating of Power for Plant Replacement of Energy Shortage

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Important parameters for operating limits are:

- CPR, Critical Power Ratio
- LHGR, Linear Heat Generation Rate

Goal: Avoid Any Clad Failures

- Clear and Worthy Goal
- Zero failures idealistic (Swedish Goal is one failure per 10^5 fuel rods)
- Dryout criterion used nearly everywhere
  - No failures because of dryout except Oskarshamn 2 which had stationary dryout
- Failures of other reasons

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Utilities' economic interests to utilize fuel more efficiently

- Optimisation of the fuel.
  - 8X8 to 10X10
  - Spacers, transients
- Dryout criteria replaced by time/temperature limit, data base developed
  - Importance dependent on reactor type (Most for ABB Internal pump, least for GE)
- Improvements in core monitoring system
- Direct-fixed and delayed-gliding APRM (Power control by pumps, select rod insertion)
- Reclassification of certain fast transients

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BWR Reactor Type	Internal Pumps	External Pumps plus Jet Pumps	External Pumps only
Operating Limits	- LHGR - MCPR (Power, Flow Number of Pumps)	- LHGR (Power, Flow) - MCPR (Power, Flow)	- LHGR - MCPR (Power, Flow)
Reactor Protection Strategies, Power Reduction, Scram Signals	- Gliding APRM, delayed power reduction - Gliding APRM, direct power reduction - Gliding APRM, delayed thermal power scram - Gliding APRM direct neutron flux scram - Scram from steam-line valve closure - Pressure scram - Water level scram	- Gliding APRM, delayed thermal power scram or direct neutron flux scram - Fixed APRM neutron flux scram - Scram from steam-line valve closure - Pressure scram - Water level scram	- Gliding APRM, delayed power reduction - Gliding APRM direct power reduction - Gliding APRM, delayed thermal power scram - Gliding APRM direct neutron flux scram - Scram from steam-line valve closure - Pressure scram - Water level scram
Cladding Integrity Safety Limit	- No Dryout, CPR ca. 1.06 - Short Term Dryout Cladding Temp. Limit (Proposed)	- No Dryout, CPR ca. 1.06	- No Dryout, CPR ca. 1.06
Typical Operating Limits	- LHGR = 14.4 kW/ft - CPR = 1.30 at 100 % Power/Flow	- LHGR = 14.4 kW/ft - CPR = 1.20 at 100 % Power/Flow	- LHGR = 13.4 kW/ft - CPR = 1.25 at 150 % Power/Flow
Limiting Transients	- Pump Trip (ABB) - Pressurization Transient plus 1st Scram Trip Failed (KWU)	- LHGR from LOCA (SAFE/REFLOOD analyses) - No Dominant Transients for CPR Limits	- LHGR from LOCA

Table 1 A Comparison of Reactor Types, Protection Strategies and Controlling Limits

	CPR	LHGR	
	CLAD INTEGRITY	FUEL DESIGN	LOCA INITIAL CONDS.
1. PHYSICAL CONSTRAINTS	BOILING TRANSITION CPR = 1.0	FUEL / CLADDING THERMAL / MECHANICAL LIMITS e.g. PLASTIC STRAIN, CENTRE-MELT	NO CORE DEGRADATION 10 CFR : 50.46 2200 °F, 1200 °C OXIDATION LIMITS
2. UNCERTAINTIES - CORE - TOLERANCES - GEXL	SAFETY LIMIT SL-MCPR 99.9 % RODS AVOID CPR = 1.0	-	-
3. BIASES FOR TRANSIENTS AND MODEL UNCERTAINTIES	Δ-CPR	Δ-LHGR -TRANSIENTS - BOUNDING MULTIPLE LIMITS	REALISTIC LOCA ANALYSES PLUS STATISTICAL UNCERTAINTIES, OR CONSERVATIVE ANALYSES AND BOUNDARY CONDITIONS
4. OPERATING LIMITS	OL-MCPR	OL-LHGR OL-MAPLHGR	OL-LHGR OL-MAPLHGR

Fig. 1 A Typical Methodology for Establishing CPR and LHGR Operating Limits

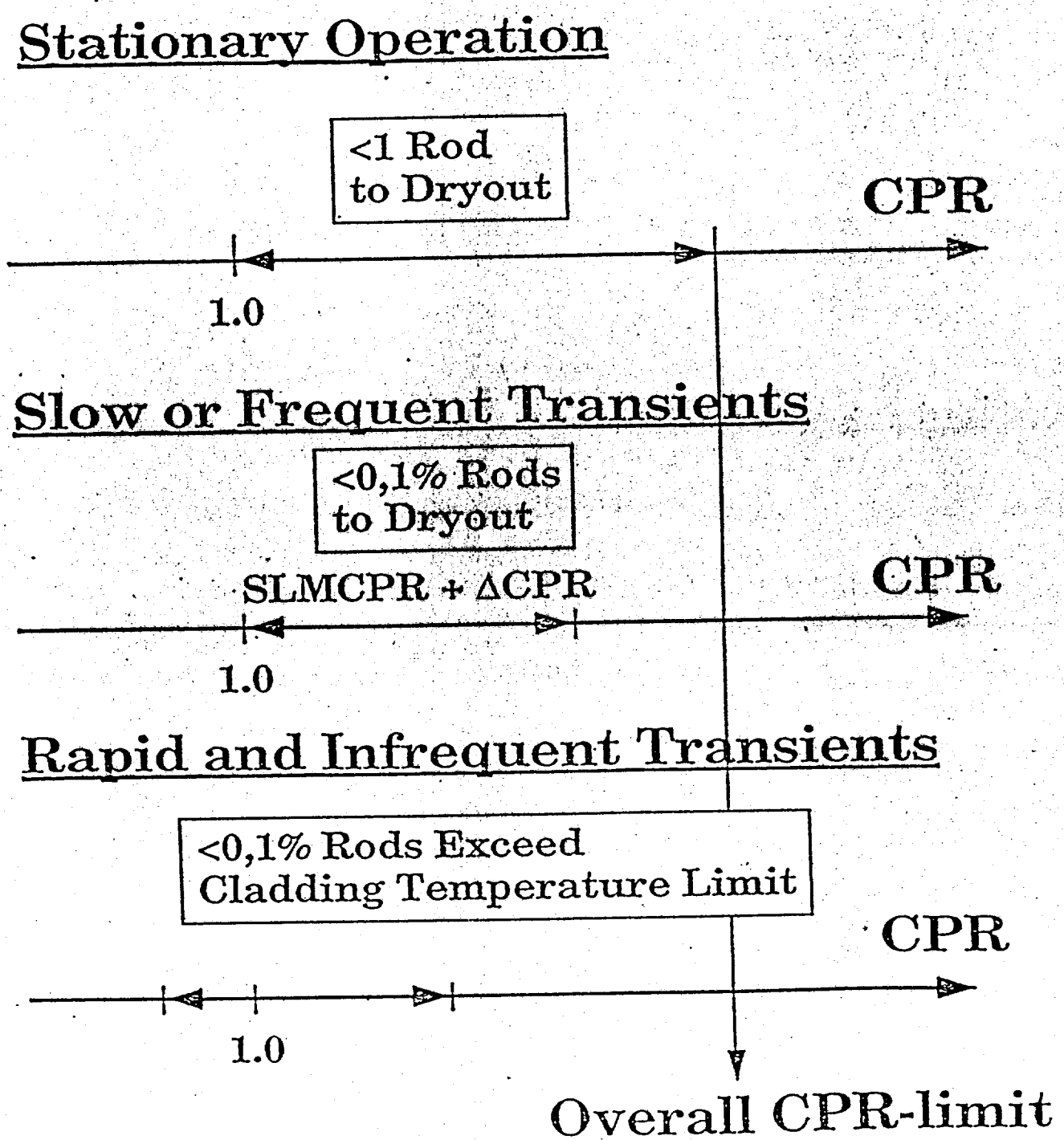

Fig. 2 A Proposal for a Revised Post-Dryout CPR Approach for Rapid, Infrequent Transients

Other areas for thermal hydraulic operational and safety limits:

- Reactivity insertion accidents
  - As a function of higher burnup
- LOCA Criteria
  - Clad embrittlement, Centerline melting, Core integrity
- Stability
  - Reactivity insertion accidents
- Pellet clad interaction
- Heat Trasfer from fuel to coolant

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Work Done on Safety Margins

Horst Glaeser

Meeting on CSNI Action Plan in the Area of Safety Margins
21-22 November 2002
Paris, France

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Presentation topics

- Safety analysis and licensing, conservative versus “best estimate”
- Quality of computer codes
- Validation of “best estimate” computer codes
- Qualification of code users
- Efforts on evaluation of uncertainties
- Activities on “best estimate” methods

Safety analysis of nuclear reactor steam supply systems

- Performed by computer simulation using complex system codes
- Margins to acceptance criteria are determined by
  - a) conservative evaluation model calculations
  - b) "best estimate" calculations supplemented by uncertainty analysis of code results

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Why do we use "best estimate" methods?

- Conservative hypotheses were introduced to cover existing uncertainties in the 1970s

- Intentional conservatisms may not lead to conservative results

  1. Example: High power during SB LOCA may lead to over-prediction of swell level

     ⇒ Over-prediction of core cooling, opposite to conservative result

  2. Example: Reduction of interfacial friction in core region

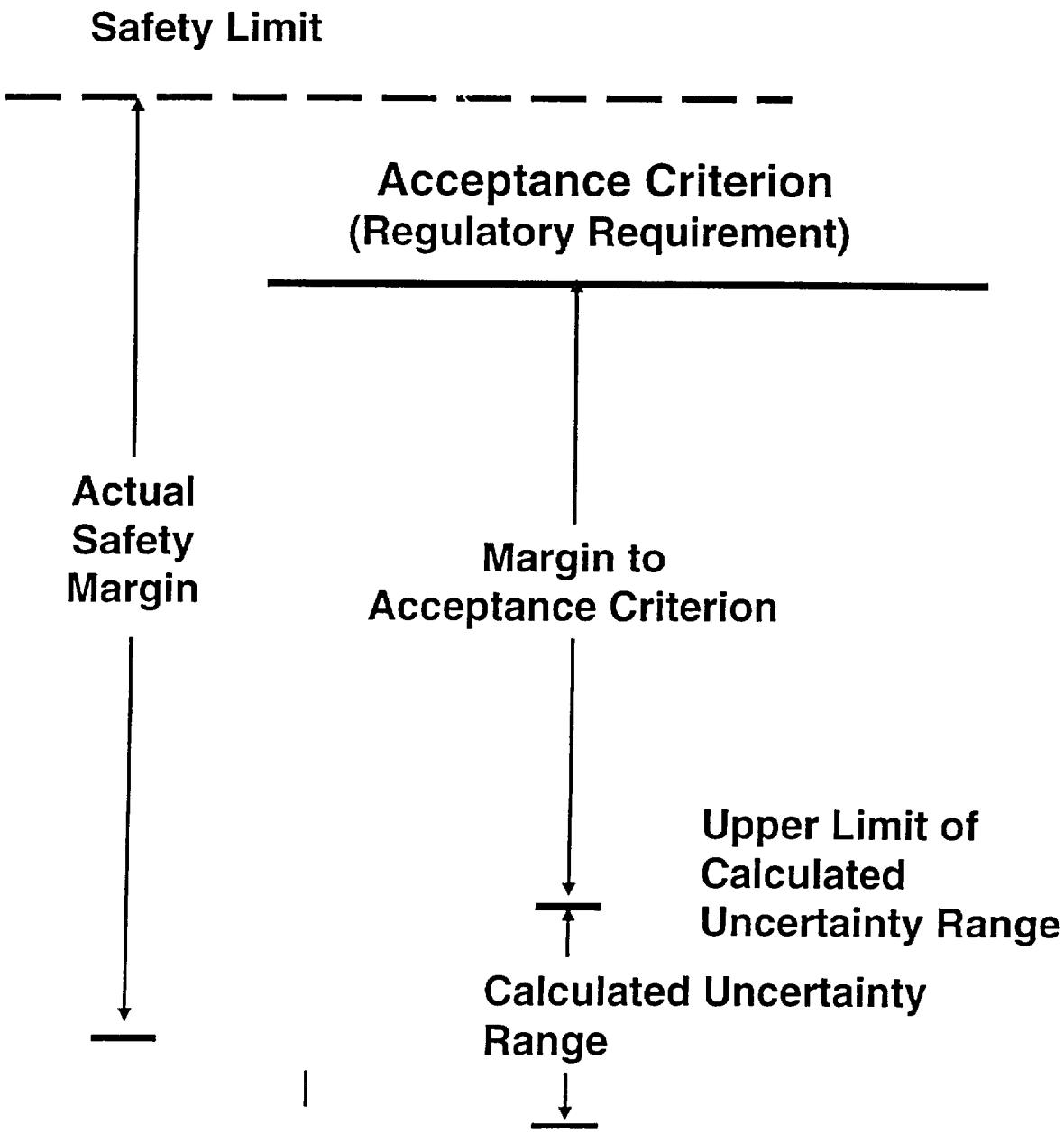
     ⇒ higher clad temperatures in upper core region

     ⇒ lower loss of water inventory, cooling earlier

Extensive international research programs improved the knowledge

- USNRC revised the LOCA rule to permit realistic analysis, with appropriate accounting for uncertainties in 1989 [10 CFR 50.46 and Appendix K; USNRC Regulatory Guide 1.157]
- The new rule and the methodologies to quantify code uncertainty established a basis for using best-estimate codes in the formal licensing process

Margin illustration



Best estimate

- Best estimate calculations: Consideration of the best available state of knowledge
- Best estimate code: Realistic models for relevant phenomena
  - Free from deliberate pessimism
  - Sufficiently detailed models to describe the relevant processes that the code is designed to model

Another trend around the world

- Reduce barriers to trade
- Countries develop market driven economics with open, competitive global trading
- Also seen in electricity supply industries
- Consequences:
  - Minimize cost of production
  - Maximize output of operating plants

  ⇒ Margin can be used to improve plant performance

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Primary regulatory acceptance criteria for LOCA design basis, valid for conservative and "best estimate" analysis

- Peak clad temperature < 1200 °C (2200 °F) to avoid brittle failures
- Maximum clad oxidation (17% of the total clad thickness before oxidation; in some countries 15%, e.g. Japan)
- Maximum hydrogen generation (1% related to total possible clad oxidation)
- Coolable geometry
- Long-term cooling
- Germany: Fuel rod failures < 10%

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Unsettled issue

- Oxidation rate of zircaloy increases significantly at burn-ups above 40 GWd/t
- Investigations to clarify the oxidation process during
  - normal operation, and
  - LOCA at higher temperatures
- Effect of these processes to the brittleness of the remaining clad material
- Different oxidation processes at different temperatures to be accounted for when comparing against the 17% LOCA limit?

Quality of computer codes (1)

- OECD/CSNI Workshop on Transient Thermal-Hydraulic and Neutronic **Codes Requirements** in Annapolis, Maryland, USA, 5-8 November 1996
  - Specific areas of code improvements identified:
    - Multi-field models
    - Interfacial area transport model
    - Multi-dimensional hydrodynamics
    - Operation at low power/ low flow
    - Operation in presence of non-condensable gases
    - 3-D neutronics
    - Low diffusive numerical methods
    - Front (steep gradient) tracking

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Quality of computer codes (2)

- OECD/CSNI Workshop on Advanced Thermal-Hydraulic and Neutronic Codes: **Current and Future Applications** in Barcelona, Spain, 10-13 April 2000
  - Long term plans for development of advanced codes
  - **Regulatory requirements for BE codes assessment**
  - **Overview of application of TH and neutronic codes for current safety issues**
  - **Needs for integral plant transients and accident analyses; known limitations**
  - Advances in next generation of TH and neutronic codes
  - Future trends in physical modelling for next generation codes
  - **Uncertainty analysis, level of confidence**
  - Simulators and fast running codes

Quality of computer codes (3)

- OECD/CSNI Workshop on Advanced Thermal-Hydraulic and Neutronic Codes: **Current and Future Applications** in Barcelona, Spain, 10-13 April 2000 (continued)

  Main results

  - Major large system TH codes have been coupled to state-of-the art 3D neutronic-kinetic codes
  - CSNI/ NSC Benchmark on PWR MSLB transient has been successfully completed, providing a demonstration of capability of coupled TH/3D neutronic codes
  - **Several uncertainty methods are available**
  - **Internal assessment of uncertainty has been demonstrated**

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Validation of "best estimate" computer codes against relevant experimental data is required

Internationally established and agreed validation matrices

- Integral tests [CSNI Report 132, March 1987; NEA/CSNI/R(96)17]
- Separate effects tests [NEA/CSNI/R(93)14/Part 1 and 2]
- VVER TH code validation matrix, preparation by OECD Support Group [NEA/CSNI/R(2001)4]
- Participation in International Standard Problems and Benchmarks
- International Code Assessment and Maintenance Program/ User Groups

Qualification of code users (1)

- User must make many choices in setting up an input deck and in running a calculation
- Experience from **International Standard Problems**:
  Large differences of code results from different code users applying the same code version and the same specifications (e.g. initial and boundary conditions, geometry, ...)
- Differences considered as **user effects**
- In other arenas, e.g. plant calculations, some of these elements may better be classified as code, scale, plant state etc. **uncertainties**
- Uncertainty analyses quantify these uncertainties and some of the user effects

Qualification of code users (2)

- User effects will be reduced by
  - Extensive user instruction, training and mentoring (most effective way)
  - User guidelines
  - Quality assurance procedures
  - Code improvements
- IAEA International Training Program for TH codes

International efforts to quantify uncertainties of code results (1)

- CSNI Workshop held on **Uncertainty Analysis Methods** in London, 1994
  - Methods presented and main speakers

• CSAU Method (USA)	Gary Wilson, INEL
• UMAE Method (Italy)	Francesco D'Auria, Univ. Pisa
• AEA Method (UK)	Tony Wickett, AEA Technology
• NE Method (UK)	Martin Trow, NNC
• GRS Method (Germany)	Horst Glaeser, GRS
• CEA Method (France)	Raymond Pochard, IPSN, et al
• Tractebel Method (Belgium)	Luc Vanhoenacker, Tractebel
• Limit Value Approach (USA)	Ed Lee, ABB-CE

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International efforts to quantify uncertainties of code results (2)

- OECD/CSNI **Uncertainty Methods Study** for Advanced Best Estimate Thermal Hydraulic Code Applications (May 1995 - June 1997)
  - Comparison of various methods for uncertainty evaluation
  - Application of methods to the same experiment: LSTF SB-CL-18 (former ISP-26)
  - 5 participants from 5 countries
    - AEA Technology, UK (ranges for parameter uncertainties)
    - University of Pisa, Italy (extrapolation of accuracy)
    - GRS, Germany (method using statistical tools)
    - IPSN, France (statistical method, similar to GRS)
    - ENUSA, Spain (statistical method, similar to GRS)

Summary of Methods Compared in this Study

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

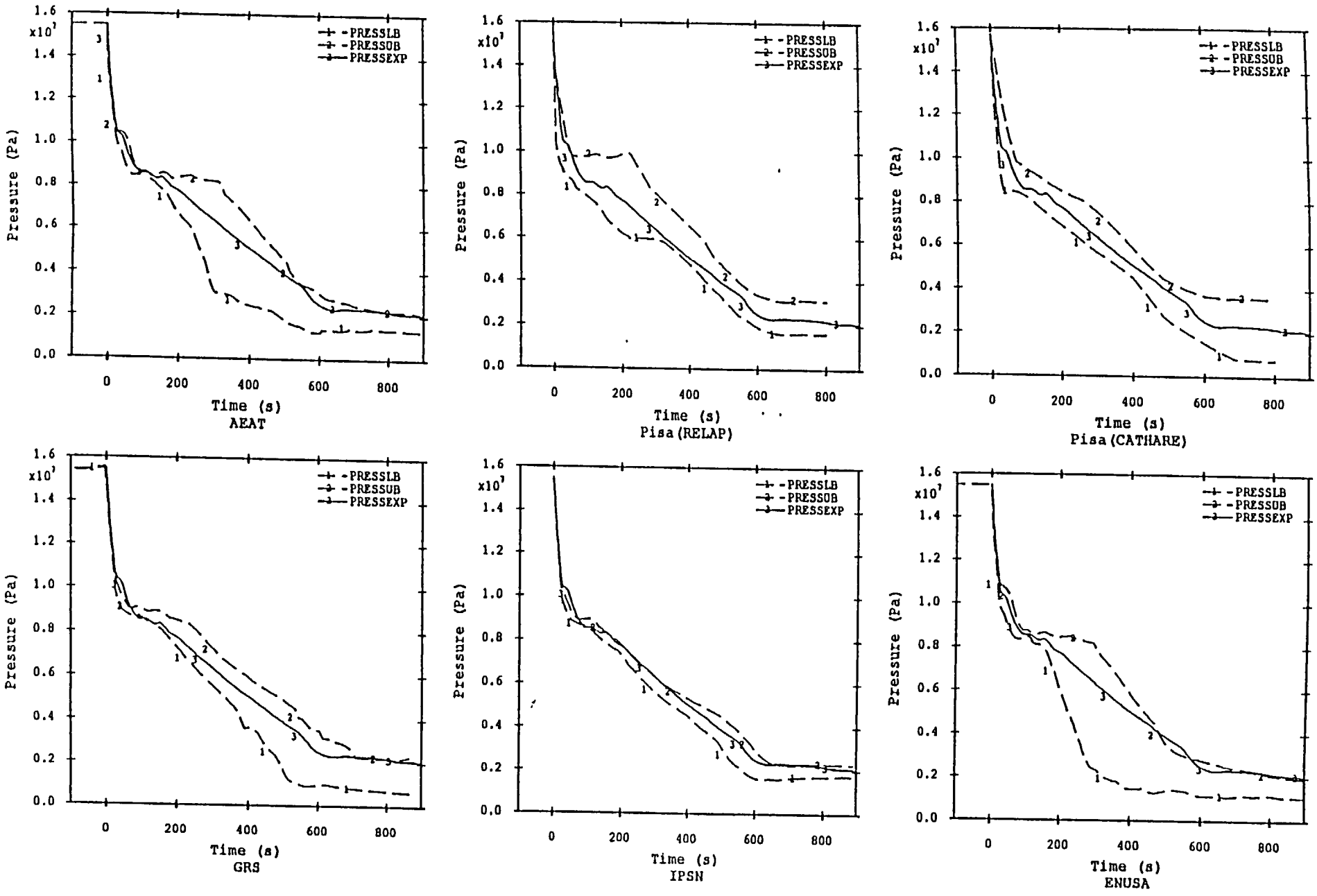

Figure 16: Uncertainty ranges and data for pressurizer pressure

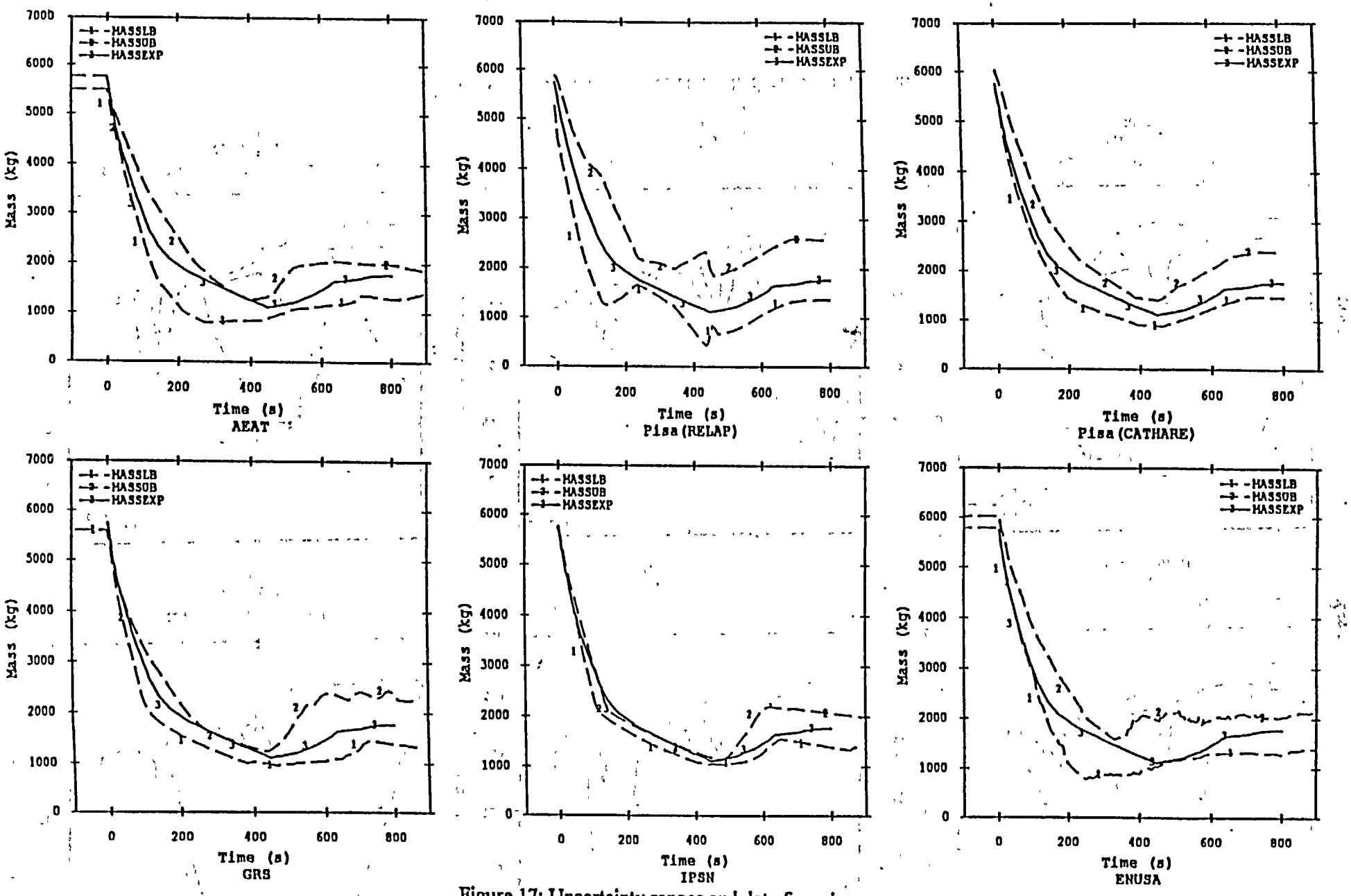

Figure 17: Uncertainty ranges and data for primary mass

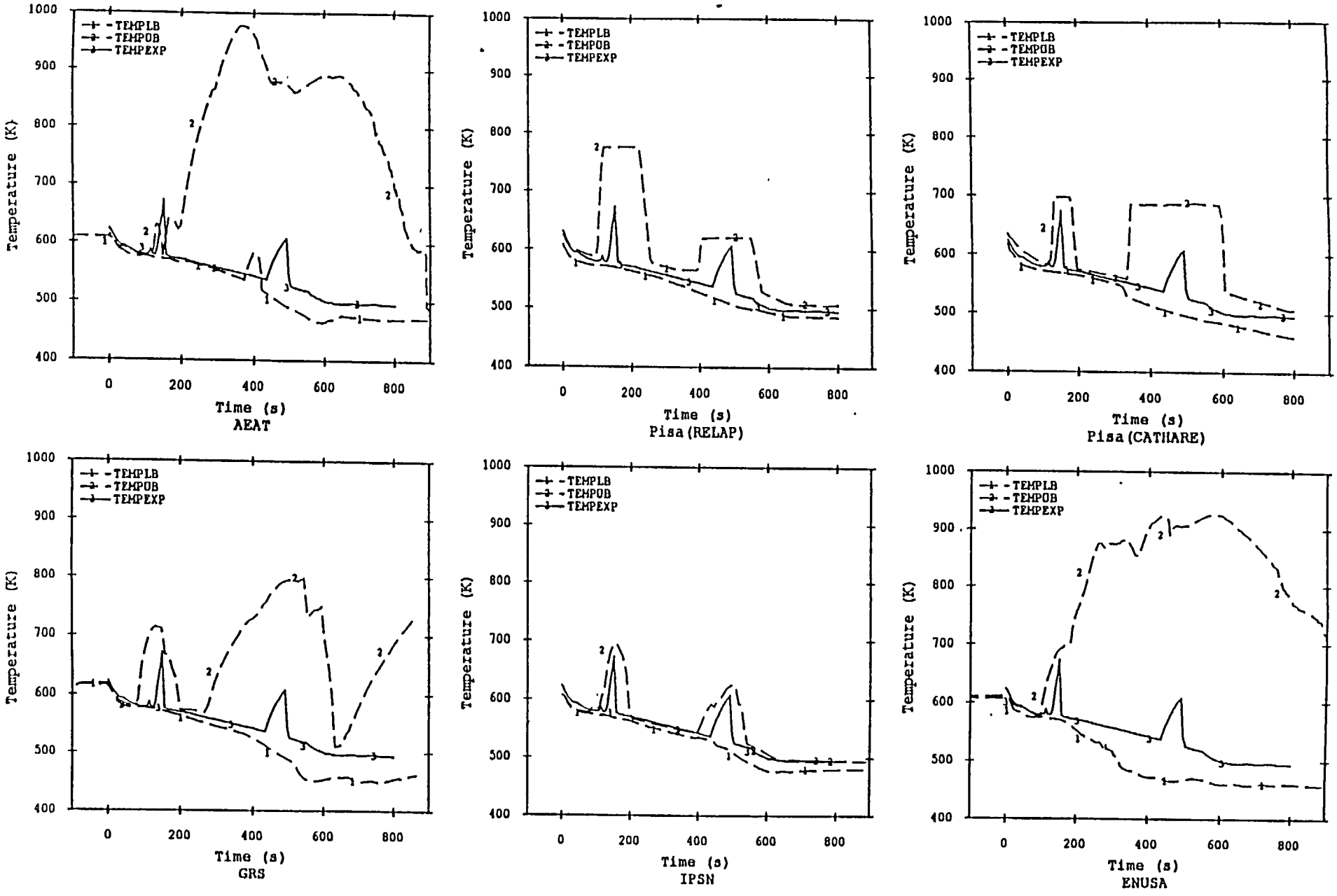

Figure 18: Uncertainty ranges and data for hot rod temperature

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Follow-on Activity after Closure of the Uncertainty Methods Study (1)

- Comparison calculations using different RELAP 5 versions (same input deck) by University of Pisa

  ⇒ RELAP 5/ MOD 3.2 calculates **170 K higher clad temperatures** compared with MOD 2 and newer version MOD 3.2.2

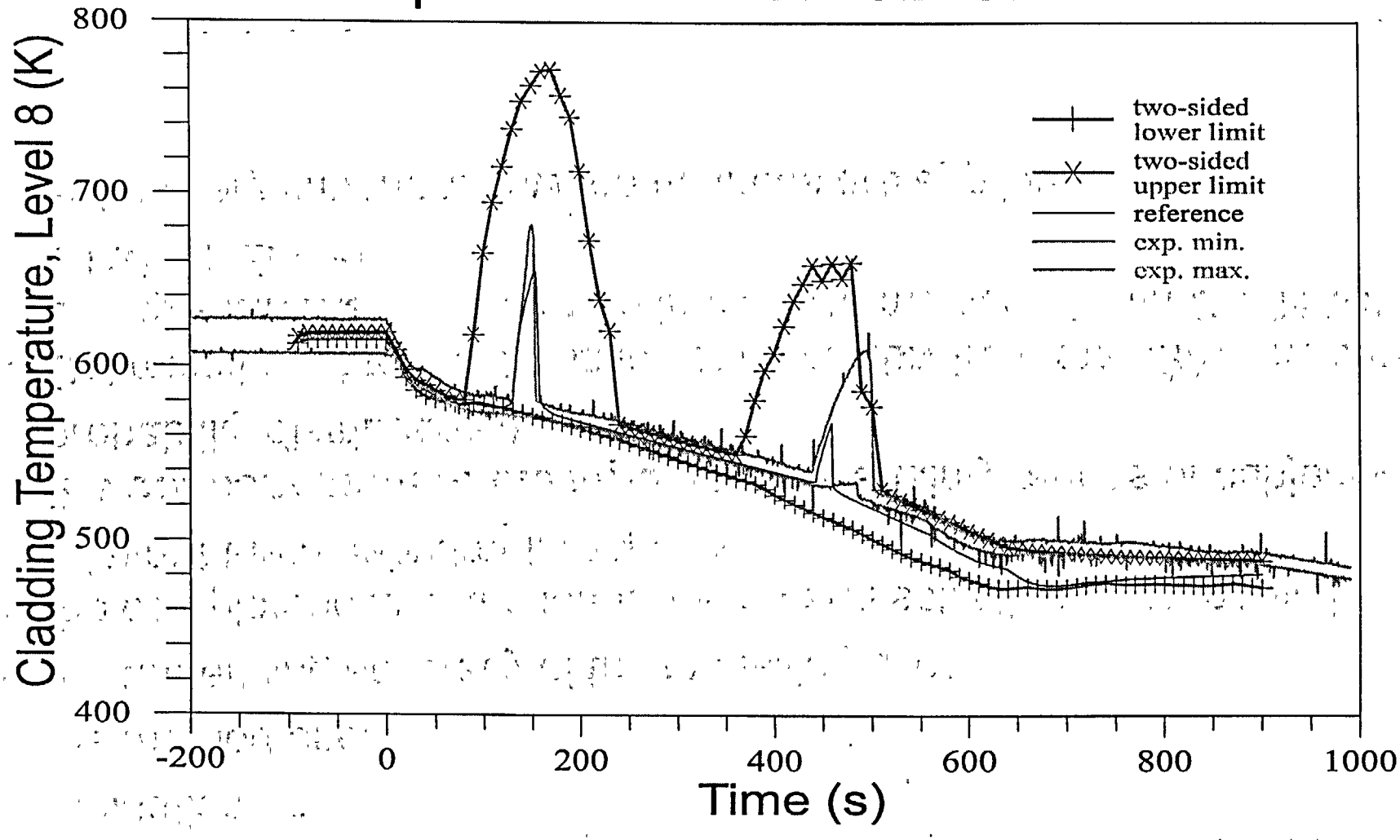
  - MOD 3.2 was used by AEAT and ENUSA
  - MOD 2 was used by University of Pisa

- This may contribute to the high upper limit of the uncertainty ranges calculated by AEAT and ENUSA

Follow-on Activity after Closure of the Uncertainty Methods Study (2)

- GRS performed new analysis with revised distributions for most important input parameters
- Lower differences of uncertainty ranges by different participants expected

Experiment LSTF-SB-CL-18



UMS comparison between the methods

The differences between the predictions of the methods came from a combination of:

- the method used
- the underlying accuracy of the modelling used
- the completeness of the identification and selection of uncertainties or relevant thermal-hydraulic aspects
- **the conservatism of the input** (eg. uncertainty ranges or subjective probability distributions)
- Optimisation of the nodalisation and possibly from the **different number of experiments investigated** in the Pisa method (5 with CATHARE, 10 with RELAP).

  ⇒ It is very important how the methods are applied

Other international activities on "best estimate" methods (3)

- OECD/CSNI Seminar on **Best Estimate Methods in Thermal Hydraulic Safety Analysis** in Ankara, Turkey, 29 June-1 July 1998
  - Several "best estimate" code applications were presented but only a few uncertainty evaluations
    - Application of a statistical uncertainty analysis in a licensing case of a new reactor (the first) by a vendor (Siemens KWU, Germany)
    - Demonstration application of the CSAU method to CANDU large break LOCA-blowdown (AECL, Canada)
    - Investigation of a PMK-2 experiment using the ATHLET code and the GRS Uncertainty Method (KFKI/ GRS, Hungary/ Germany)

Other international activities on "best estimate" methods (4)

- International Meeting on Best Estimate Methods in Nuclear Installations Safety Analysis (BE-2000), Washington, DC, USA, November 2000
  - Broad spectrum of topics
  - BE methods are as likely to increase safety as to provide economic gains
  - Issue of BE methods uncertainties not resolved yet
  - Attention should be paid to uncertainty of physical models

IAEA-Report (TECDOC) on "Safety Margins of Operating Reactors" (1)

- Technical Committee Meeting in Vienna, 15-19 October 2001 producing the report
- Contents
  - Capabilities of thermal-hydraulic computer codes including the evaluation of uncertainties
    - Current status of widely used computer codes
    - Conservative versus best estimate approach

Contents of IAEA-Report (TECDOC) on "Safety Margins of Operating Reactors" (2)

- Methods for safety margin evaluation
  - Safety limits/ margins
  - Approaches
  - Initiating events
  - Quality assurance

IAEA-Report (TECDOC) on "Safety Margins of Operating Reactors" (3)

- Utilizing safety margins in operation and modifications of NPP
  - Role of PSA and deterministic analyses in plant modification
  - Evaluation of current safety margins and identification of weak points
  - Measures to increase margins
  - Some examples
- Annex: Papers presented at the Technical Committee Meeting

GRS

IAEA-Report (TECDOC) on "Safety Margins of Operating Reactors" (4)

- Report expected to be published until end of 2002
- Contributors to drafting and review

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General comments (1)

- Considerable **interest** in uncertainty evaluation from research organisations, utilities and regulators
- Some potential users wish to improve **practicability** of methods ⇒ internal assessment of uncertainty, results from several accident scenarios needed
- **"Useful" results** from the point of view of a utility wanting to reduce conservatisms ⇒ e.g. see CSAU applications in USA
- **Differences** in uncertainty distributions due different experiments used
  - to transform validation results into input parameter distributions and ranges from separate effects experiments, or
  - to determine uncertainty of code results directly by differences between calculated and measured PCT from integral experiments

General comments (2)

- **Code uncertainty** partly evaluated directly by determining the differences between calculated and measured values of different experiments investigating the same accident scenario
  - Concern of scaling distortions of integral experiments, different time scales
  - Concern of compensating errors and error propagation (from USNRC)
  - Additional uncertainties of **plant conditions and fuel related parameters** to be quantified (additional bias)

General comments (3)

- Statistical uncertainty evaluation considered as time consuming
  - However, e.g. USNRC rules recommend a **95% (or more) probability statement** that licensing limits are not exceeded
  - Alternative: Conservative analysis applying Appendix K rules; disadvantage of Appendix K application: High level of code validation can not be utilized; intentional conservatims may not always lead to conservative results
- Some organisations prefer a bounding approach by using best estimate codes with conservative parameter values, initial and boundary conditions
- Specification of values for uncertain input parameters is necessary for bounding analysis as well as for statistical uncertainty analysis

GRS

General comments (4)

- Simplified method to be used with confidence in licensing process?

  ⇒ Bounding best estimate approach using SECY-83-472, licensed by Westinghouse and General Electric no longer allowed in USA

- Definitions and nomenclature need to be fixed

- Not clear that consensus on creation of an "effective" standard for BE methods and applications is possible

- Regulatory guidance for uncertainty analysis wanted

GRS

Conclusions

- Great deal of communality of practices of safety analysis to determine safety margins in different countries, details may differ:
  - "Best estimate" calculations supplemented by uncertainty statement of the code results are preferred versus
  - Conservative evaluation model calculations
- Method to evaluate uncertainties in licensing procedure are not yet settled in most countries
- Probabilistic uncertainty methods need to be accepted by regulation
- Effort towards harmonization of best estimate analysis and uncertainty methods and their application for regulatory purposes would be worthwhile

International situation in licensing (1)

- Regulations in most countries permit the use of best-estimate codes [CSNI Status Summary on Utilization of Best-Estimate Methodology in Safety Analysis and Licensing; NEA/CSNI/R(96)19, October 1996]
- For example, best estimate method including uncertainty evaluation was applied for
  - Netherlands: NPP upgrade renewal license of the Dodewaard nuclear power plant, performed by General Electric, reviewed by GRS
  - USA: Updates to the Final Safety Analysis Reports of approximately twenty plants,
    performed by Westinghouse using the Code Scaling, Applicability and Uncertainty (CSAU) method and response surfaces;
    main concerns of USNRC were to investigate compensating code errors, propagation of uncertainties and scaleability

International situation in licensing (2)

- USA: An AP 600 LB LOCA analysis, performed by Westinghouse using the CSAU method
- Brazil: Angra-2 NPP licensing analysis in Brazil, performed by Siemens (now Framatome ANP), reviewed by GRS; first application in a licensing process of a new NPP
- France: Utility EDF presented statistical method (95% probability statement) using a response surface to substitute the BE code (CATHARE-GB), plus a deterministic "realistic" calculation enveloping the 95% values.
  IRSN reviewed this method for the French Safety Authority. The response surface was not accepted, but the envelope based on representative integral tests was.
  IRSN uses the Wilks' formula to evaluate the uncertainty relying on actual code results without approximations by fitted response surfaces similar to the GRS method.

International situation in licensing (3)

- Canada: Guidelines and applications by Ontario Power Generation and AECL, under review by an international expert panel
- Korea: The KEPCO (Korea Electric Power Corporation) Realistic Evaluation Methodology for power increase (in any case, license renewal every 10 years) follows essentially the CSAU method, however, using the Wilks' formula (proposed by GRS), performing 59 computer code runs to obtain 95% probability statements
- Lithuania: Lithuanian Energy Institute performs uncertainty analyses for Ignalina Unit 2 NPP (RBMK-1500) licensing process using the GRS method (e.g. group distribution header blocking event)
- Germany: Efforts are underway to include realistic analyses and uncertainty evaluation in the licensing regulation.

CSN CONSEJO DE SEGURIDAD NUCLEAR

Safety margins: deterministic and probabilistic views

Javier Hortal - Nuclear Safety Council - Spain

CSN

Purpose and context

- The concept of safety margin is used in very different ways.
- It is a basic element of the safety analysis.
- Safety analysis is a multidisciplinary task whose goal is the design or the assessment of the protection of a plant.
- It is necessary to define a reference framework for inter-discipline communication.
- A general concept of safety margin can be derived from very basic premises. Any practical implementation of the concept should fit in the general concept.

CSN

Definitions

Damage: Undesired consequence of the behaviour (expected or not) of a plant.

Plant States:

Logical state: Set of discrete variables describing the states of the plant systems: $j = \{j_1, j_2, ...j_N\}$

Process vector: Set of real variables describing the plant process evolutions: $\vec{x}$

Plant dynamics for logical state j: Set of functions $\vec{g}_j$ that describe the evolution of the process vector from initial conditions $\vec{x}_0$, i.e., $\vec{x} = \vec{g}_j(t, \vec{x}_0)$.

Event: Instantaneous change in the logical state j, occuring at time t, denoted by E^t. In the general case, an event will change the plant dynamics.

CSN

Definitions

Transient: Evolution of the logical state j and the process vector $\vec{x}$ from an initiating event E^0 occuring at steady state, until new steady state is reached. (Steady state can be considered as a null transient).

Damage variable: Function of the process vector $D_i(\vec{x}, t)$ that quantifies the generation of an undesired effect or the proximity to that effect. There are **many possible** damage variables.

Risk: Relationship between amount of damage and likelyhood (frequency).

The concept of risk is essentially bidimensional and associated to a damage variable. The acceptability of a damage depends on its expected frequency.

CSN

Damage limit

- Given D_i, the damage limit is a function that defines the maximum acceptable damage for each frequency ν, i.e., $D_i^L = D_i^L(\nu)$
- It can be represented in the risk plane of the damage variable D_i
- Main attributes of the damage limit:

  -- Monotonically decreasing function.
  -- Vertical limit or asymptote on a low damage value.
  -- Horizontal limit or asymptote on a low frequency value.

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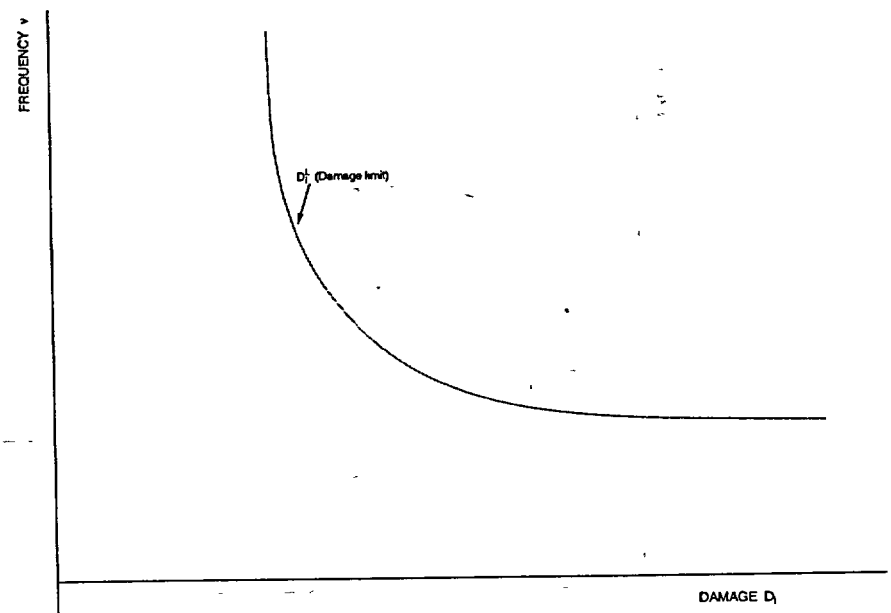
Damage limit



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Characterisation of the plant safety

- Damage is generated during transients
- A transient, denoted by q can be viewed as:

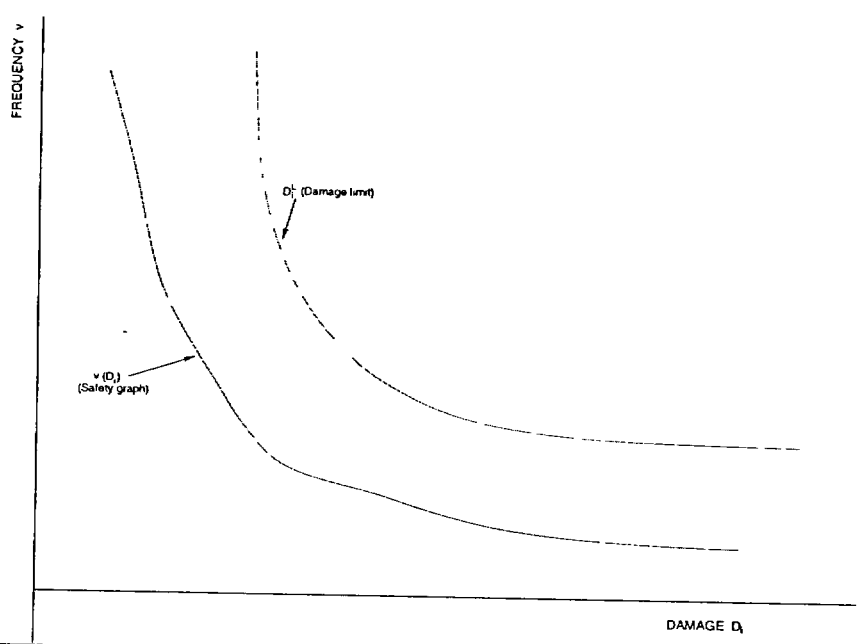
  -- A random process characterised by its mean frequency ν_q.
  -- A dynamic history that genarates an amount of damage D_{iq}.

- A damage of magnitude D_i occurs when a transient generating $D_{iq} \geq D_i$ occurs.
- Plant safety graph:

$$\nu(D_i) = \sum_{(q|D_{iq} \geq D_i)} \nu_q$$

CSN

Characterisation of the plant safety



CSN

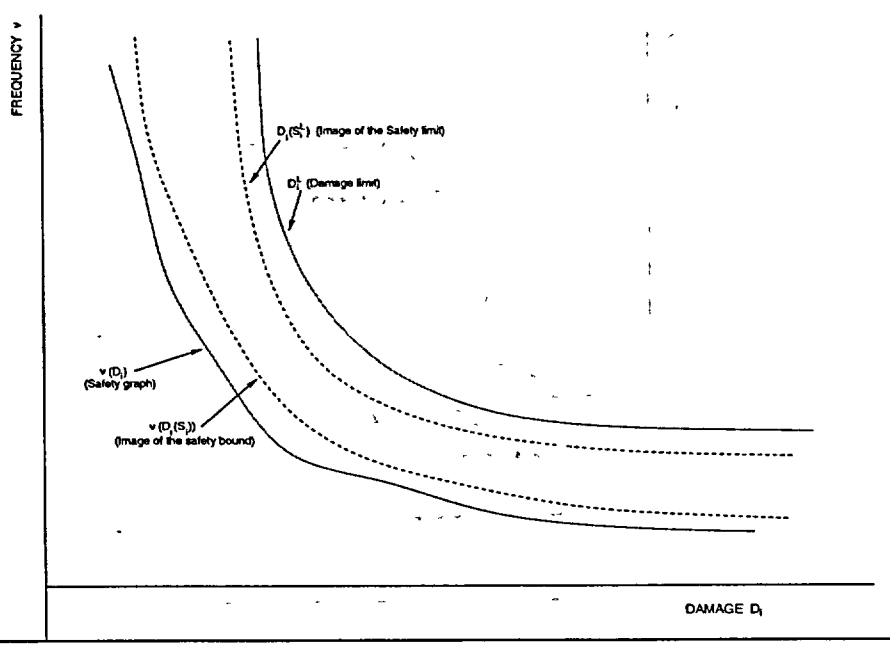
Safety variables and safety limits

- Safety variable: intermediate variable between damage and process variables. Used when the damage variable is unpractical.
- A safety variable S_i can be associated to damage variable D_i if there is a function $\mathbf{D_I}$ such that: $\mathbf{D_I}(S_i(\vec{x}, t)) \geq D_i(\vec{x}, t)$.
- A safety limit $S_i^L = S_i^L(\nu)$ can be imposed to the safety variable if the bounding condition: $\mathbf{D_I}(S_i^L(\nu)) \leq D_i^L(\nu)$ is met. The safety limit represents a **necessary condition** for damage.
- A new figure of merit of the plant safety can be defined in terms of the safety variable:

$$\nu(S_i) = \sum_{(q|S_{iq} \succeq S_i)} \nu_q$$

CSN

Characterisation of the plant safety



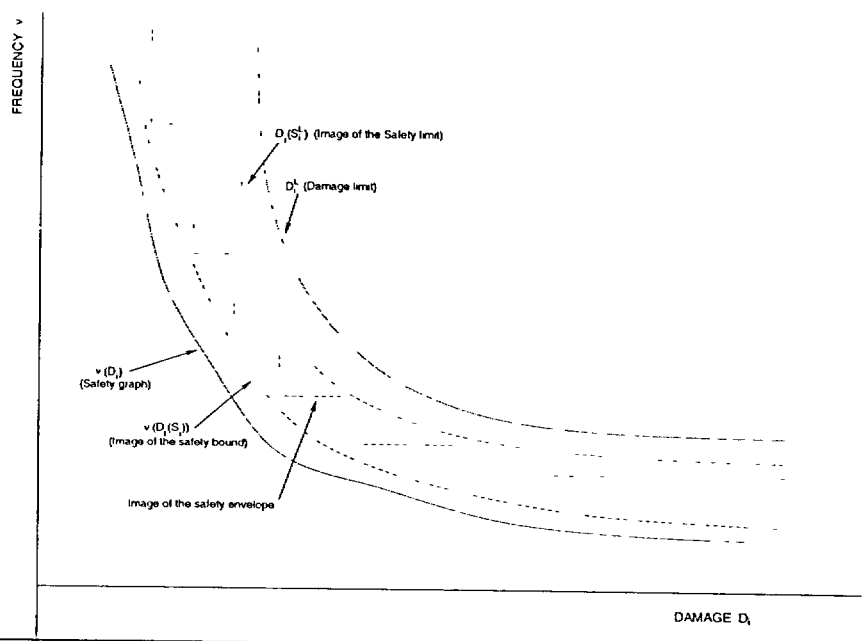
CSN

Grouping and enveloping transients

- Plant safety graphs cannot be calculated due to the unlimited number of possible transients.
- Practical approach: grouping of transients.
- Grouping process must generate an envelope of the safety graphs in order to preserve safety.
- Group Q attributes:
  - -- Collective frequency: $\nu_Q = \sum_{q \in Q} \nu_q$
  - -- Bounding value of damage: $D_{\imath Q} \geq D_{\imath q} \ \forall q \in Q$
  - -- Bounding value of safety variable: $S_{\imath Q} \succeq S_{\imath q} \ \forall q \in Q$
- Safety envelope: $\widehat{v(S_\imath)} = \sum_{(Q|S_{\imath Q} \succeq S_\imath)} \nu_Q$. It is an envelope since $\widehat{v(S_\imath)} \geq \nu(S_\imath)$

CSN

Characterisation of the plant safety



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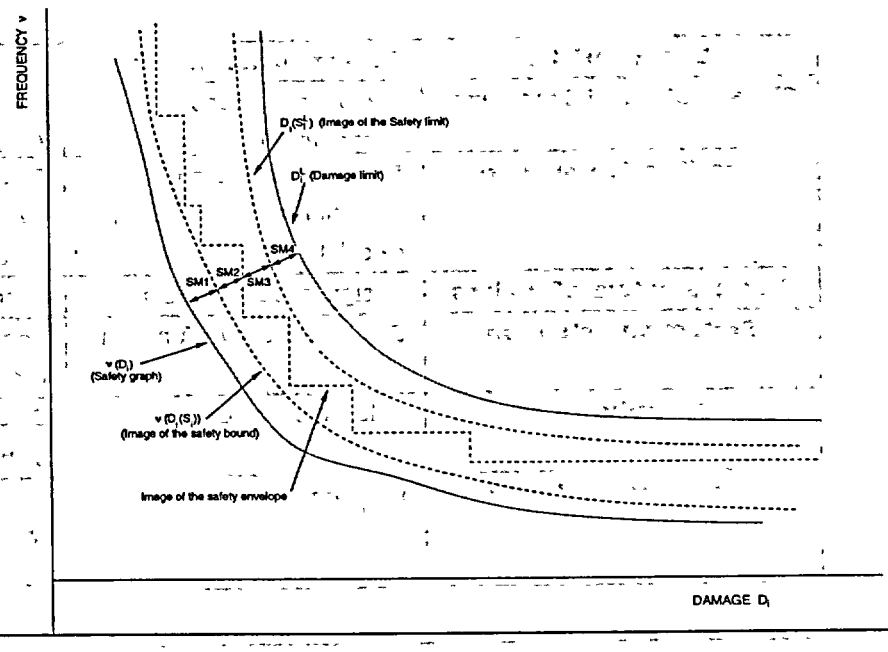
The concept of safety margin

- Global safety margin: <<distance>> between safety graph and damage limit
- Partial safety margins:

SM1 <<Distance>> between safety graph and safety bound.
SM2 <<Distance>> between safety bound and safety envelope.
SM3 <<Distance>> between safety envelope and safety limit.
SM4 <<Distance>> between safety limit and damage limit.

CSN

Characterisation of the plant safety



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Steps of the safety analysis

A complete safety analysis consist of:

- Determination of the damage variable D_i and the damage limit $D_i^L(\nu)$ (or freq. limit $\nu^L(D_i)$).
- Determination of the safety variable S_i and the corresponding function D_I. (Safety margin SM1).
- Determination of the safety limit $S_i^L(\nu)$ (or freq. limit $\nu^L(S_i)$). (Safety margin SM4).
- Determination of the safety envelope: grouping of transients. (Safety margin SM2).
- Demonstration of the safety margin SM3.

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Practical approaches to safety analysis

Characteristic	Deterministic Analysis	Probabilistic Analysis
Classification of transients	Frequency groups (*Conditions*)	Damage groups (*success/damage*)
Classification support	Implicit or explicit assumptions. No probability calculation	Estimation of damage from event combination. Few confirmatory calc.
Representative of a class	Design basis transient	Event tree sequence
Protection actuation	Single protective function	Combinations of failed and successful protective functions
Quantification	Calculation of damage	Calculation of frequency (probability)
Acceptance criterion	Damage below limit	Frequency below limit (No acceptability limit is established)

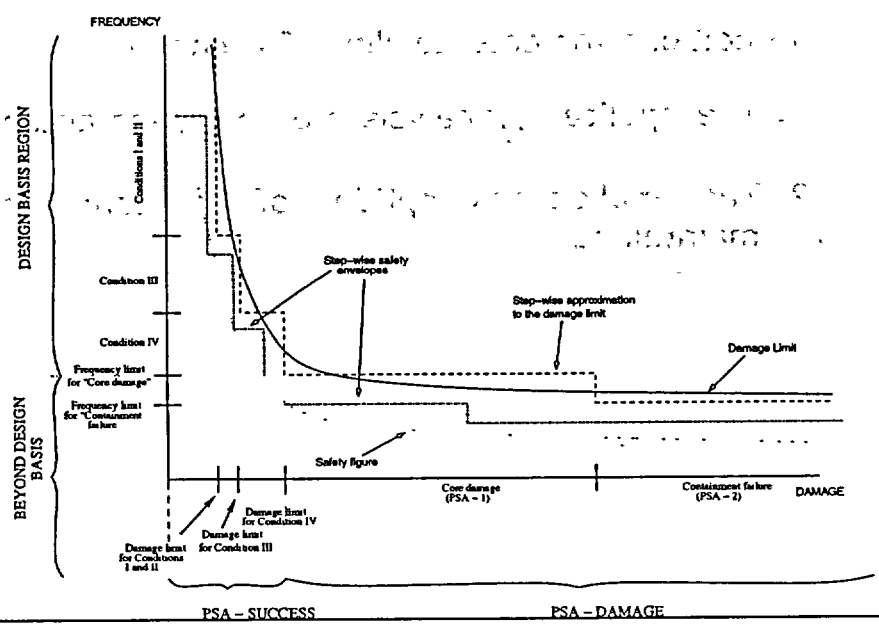
CSN

Practical approaches to safety analysis

Steps of the analysis	Deterministic Analysis	Probabilistic Analysis
Step 1: Damage variables and limits	Defined in applicable standards. Example: ANSI-51.1/ANSI-N 18.2	Only one damage variable: CD No $\nu^L(D_i)$ defined.
Step 2: Safety variables and SM1	Mostly generic design studies. Example: DNBR for C-I and C-II	Cladding temperature or loss of long term cooling.
Step 3: Safety limit and SM3	Generic or specific studies. Example: minimum DNBR for C-I and C-II	No $\nu^L(S_i)$ defined. No SM3.
Step 4: Grouping, enveloping and SM4	Double grouping: DBTs and *Conditions*. Completeness: selection of the DBTs. Maximum damage for each *Condition*	Event tree delineation. Damage/success classification. Bounding frequencies
Step 5: Demonstration of SM3	Acceptable resuts of DBT.	No SM3 defined. Criteria on CDF increments

CSN

Unified view of the safety analysis



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Consistency and compatibility of SA methods

- Separate fields of application
- Mutual benefit:
  - -- Probabilistic techniques may be used to assess reliability assumptions in deterministic methods.
  - -- Simulation techniques may be used to assess event tree delineation.
- Interactions: changes in one side may invalidate implicit assumptions in the other:
  - -- Change of a failure probability may invalidate SFC.
  - -- Change of a setpoint may invalidate ET delineation.

CSN

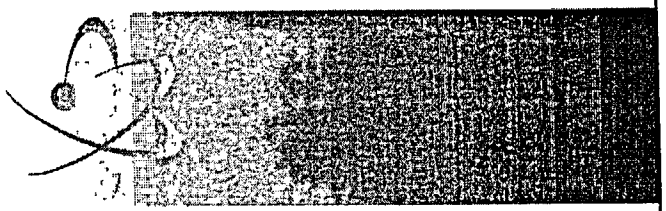
Regulatory requirements

Conditions for acceptability of a change:

1. The applicable analysis method gives acceptable results.
2. The implicit assumptions of the complementary method are maintained.

Canadian Nuclear Safety Commission / Commission canadienne de sûreté nucléaire

Assessment of Safety Margins in the Canadian Nuclear Industry



M.A. El-Hawary
Canadian Nuclear Safety Commission
November 2002
Canada

Canadian Nuclear Safety Commission / Commission canadienne de sûreté nucléaire

Best Estimate and Analysis and Uncertainty (BEAU) Methodology

- The Canadian utilities and AECL have developed BEAU methodology in the last three years. Some licensing submissions have been made using BEAU on a trial basis.
- An independent international Expert Panel has reviewed and commented on the Canadian BEAU methodology and its applications (June 2002).

Canada

Canadian Nuclear Safety Commission / Commission canadienne de sûreté nucléaire

Analyses Submitted Using BEAU

- Pump trip for Darlington NGS. The analysis led to modifications to reactor trip parameters;
- LLOCA analysis for Darlington, Bruce, and CANDU-6 NGS's;
- LLOCA analysis for Gentilly-2 NGS fuelled with CANFLEX bundles;
- Channel flow blockage and pump seizure events for MAPLE research reactor.

Canada

Canadian Nuclear Safety Commission / Commission canadienne de sûreté nucléaire

The Need for Safety Margins Evaluations

1. Request for refurbishment/life extension of Point Lepreau & Gentilly-2 NGS's;
2. Plant ageing:
   - *Fuel Channels:*
     PT creep, delayed hydride cracking & changes in material properties;
   - *Boilers:*
     Stress corrosion cracking, inter-granular attacks, fretting & pit growth;
   - *Feeders:*
     Flow-assisted corrosion & steam induced stress corrosion cracking.

Canada

Canadian Nuclear Safety Commission / Commission canadienne de sûreté nucléaire

The Need for Safety Margins Evaluations (Cont'd)

3. New Fuel Designs:
   - *CANFLEX fuel bundle:*
     43 elements, natural uranium, higher CHF & slightly higher void reactivity coefficient;
   - *Low Void Reactivity Fuel:*
     43 elements, enriched uranium with center poison, lower void reactivity coefficient & higher CHF.

Canada

Canadian Nuclear Safety Commission / Commission canadienne de sûreté nucléaire

Regulatory Activities

1. Generic Action Item on computer code validation
2. Generic Action Item on physics codes update.
3. Support BEAU methodology.
   - Encourage the Industry to develop BEAU methodology;
   - Participated with the Industry in BEAU Expert Panel review;
   - Developed a draft guide for BEAU;
   - Developed regulator's expectations of the methodology. Focus is placed on code validation, identification of important phenomena and ranking of input parameters;
   - Plan to participate with the Industry in discussions to incorporate BEAU in licensing and in regulatory decision making

Canada

Canadian Nuclear Safety Commission / Commission canadienne de sûreté nucléaire

OPG Plans for LLOCA Margin Restoration

- Use BEAU analysis to supplement the existing Limit of Operation Envelop (LEO) analysis in the short term;
- Use BEAU analysis to evaluate LLOCA safety margins;
- Develop risk-informed approach for licensing submissions, in the long term, with BEAU as a central element;
- Improve LLOCA margins by making design changes (e.g , trip timing, fuel design and others).

Canada

OECD - NEA
Committee on the Safety of Nuclear Installations

Meeting to Develop a CSNI Action Plan
in the Area of Safety Margins

NEA Headquarters, Le Montparnasse, Issy-les-Moulineaux
28 - 29 October 2002

PRESENTATION FOR ROUND TABLE DISCUSSION

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

Djordje Vojnovic
Slovenian Nuclear Safety Administration (SNSA)

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

PRESENTATION CONTENT

- SAFETY MARGINS - INTRODUCTION
- SAFETY MARGINS MANAGEMENT

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

SAFETY MARGINS (SM) - INTRODUCTION:

- Safety margins (for operating plants) are result of the conservative design approach reflecting the state of knowledge and perception of risk/safety at the time of design,
- Various types of SM in various areas of design and operation principles and practices:
  - Systems/Structures/Components (SSC) - overdesign, overcapacity
  - time frame - duration and sequencing,
  - material - parameters and properties,
  - programmatic aspects - way of operation, limiting cond. for oper., procedures, etc.,
- Changes of build-in SM through plant life:
  - material degradation,
  - plant modifications,
  - environment changes (natural, man-induced), etc.

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

SAFETY MARGINS (SM) - INTRODUCTION (Cont.):

- Parameters used for indicating the level of SM:
  - relation between the designed and operating and/or design basis accident parameters, including the level of uncertainties - (conservative and best estimate values !)
    - capacity/capability of various SSCs,
    - etc.
  - time frame,
  - balance of prevention and mitigation,
  - level of available redundancy (at certain point of time), etc.

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

SAFETY MARGINS MANAGEMENT (Cont.):

- Needs for SM management:
  - state of knowledge,
  - economic aspects,
  - plant (material) ageing,
  - operating experience,
- Safety margins changes - identification and assessment:
  - power uprates,
  - operating cycles duration,
  - PSR - plant safety assessment regarding the current safety standards,
  - plant modernisation and modifications projects,
  - different applications of PSA,
  - transition to RIR,
  - events and OEFP, etc.

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

SAFETY MARGINS MANAGEMENT (Cont.):

- The main premise of the SM management is the reassessment of the "remained safety factor" taking into account two (coupled) basic categories:
  - uncertainties,
  - acceptance criteria,
- Acceptance criteria shall incorporate variety of influencing factors,
- Plant modifications:
  - cumulative effects of changes,
  - tracking of the changes,

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

ROLE OF PSA

- Quantification of SM - reflection of changes in PSA analysis/results ?
  - It is expected that the PSA can be used for quantifying the SM,
  - Assessment of the plant changes impact on PSA results do not mean that its involve SM changes quantification,
  - Dedicated PSA re-analysis is necessary in some cases,

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

NATIONAL ACTIVITIES:

- There is/was no activities dedicated to identify and manage the SM,
- Through the safety assessment/analysis of some plant changes there were some re-evaluation which could be considered also as analysis of some aspects of SM:
  - Boron Injection Tank (BIT) concentration reduction:
    - PSA assessment - changes in the MLOCA event tree:
      - new (split) sequences in event tree,
      - changes in the ECCS recirculation timeframe,

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

Plant modernisation - some examples of SM changes:

- Power uprate and consequently changed operating parameter,
- New SG with higher volume and bigger heat transfer surface,
- Repeated safety analysis - shorter response times in general,
- New more conservative decay heat curve,
- Use of LBB approach and the new more conservative approach for reactor vessel internal load calculation,
- Different methodology for seismic loads calculation,

Impact on PSA results - see the next viewgraphs

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

Impact of plant up-rate and plant modifications on PSA results:

Modifications	Level 1	PDS	Level 2	IPEEE
1 Uprating : power increasing of 6.3% (1882MWt => 2000MWt)	Impact	No impact ( )	Not expected	Impact
1.1 Impellers of the FW pumps changed	No impact	No impact	No impact	No impact
1.2 MFW Control Valves internal changed to increase Cv	No impact	No impact	No impact	No impact
1.3 (3/3) Condensate pumps or 3 FPs running instead of (2/3)	No impact	No impact	No impact	No impact
NB 1 3 CPs are preferred	-	-	-	-
NB.2 possibility of 3 CPs and 3 FPs for certain pressures	-	-	-	-
1.4 #2 low pressure heater drain line increased	No impact	No impact	No impact	No impact
1.5 Trim of the AFW Control Valve replaced	No impact	No impact	No impact	No impact
1.6 The thermal hydraulic parameters are governed by the operating window	Impact	No impact	Not expected	Impact
2 SAT removal	No impact	No impact	No impact	Impact
3 SG replacement : Model D4 replaced by KWU SG with feed ring	Impact	Numerical impact	Numerical impact	Impact
3.1 Replacement itself	Impact	Impact	Impact	Impact
3.2 Pre-heater and its protection logic are removed	No impact	No impact	No impact	No impact
3.3 Main Feedwater modifications	Impact	Impact	Impact	Impact
3.3.1 Portion of pipe added to relocalize the inlet of the MFW nozzle	No impact	No impact	No impact	No impact
3.3.2 Bypass line removed except for the AFW line	Impact	Impact	Impact	Impact
3.3.3 Valves to regulate the split feeding removed	No impact	No impact	No impact	No impact
3.4 Flow alarm value increased to the total flow	No impact	No impact	No impact	No impact
3.5 Blowdown lines taking suction from the downcomer removed	No impact	No impact	No impact	No impact
3.5.1 System to adjust the flows tubes/downcomer removed	No impact	No impact	No impact	No impact
3.6 New flow control system : Median Signal Selector	No impact	No impact	No impact	No impact
3.6.1 A third measurement line added if only two are present	No impact	No impact	No impact	No impact

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

Impact of plant up-rate and plant modifications on PSA results:

Modifications	Level 1	PDS	Level 2	IPEEE
4. (90%) of the snubbers removed	No impact	No impact	No impact	Impact
4 1 Some are replaced by limit stops in auxiliary lines, other are removed	No impact	No impact	No impact	Impact
4 2 (3/5) of the snubbers around the SGs are remaining discussion with safety authorities is still in progress	No impact	No impact	No impact	Impact
4 3.1 Three little oil tanks instead of one big tank if 2 are removed	No impact	No impact	No impact	No impact
5 Inadequate Core Cooling System (ICCMS) implementation	Impact	No impact	No impact	Impact
5 1 Core Exit Thermocouple Monitoring System (CETMS)	Impact (<)	No impact	No impact	Impact (<)
5 2 Reactor Vessel Level Instrumentation System (RVLIS)	Impact	No impact	No impact	Impact
5 3 Sub-cooling Margin Monitoring System (SMMS)	Impact (<)	No impact	No impact	Impact (<)
6 Leak Before Break (LBB) Concept	Impact	Impact	Impact	No impact

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

Impact of LBB concept on CDF:

LLOCA Frequency (yr-1)	Core Damage Frequency due to LLOCA (yr-1)	Total Core Melt Frequency (yr-1)	Contribution to Core Damage Frequency (%)
5.0E-04	1 8665E-06	5.396E-05	3 46
8 0E-06	2.9864E-08	5 212E-05	0 06

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

Impact of LBB concept:

Level 1 IPE	Plant Damage States	Level 2 IPE	IPEEE
Impact • LLOCA initiating event frequency reduced (refined specific evaluation required) • Core damage frequency due to LLOCA reduced • Total core damage frequency reduced by 4%	Impact • PDS frequency associated with "A" initiator reduced, but the order of magnitude is maintained	Impact • Containment release frequency reduced, but the order of magnitude is maintained	No impact

NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

SG replacement impacts:

Level 1 IPE	Plant Damage States	Level 2 IPE	IPEEE
Impact • Initiating event frequencies impacted • HRA related to MF# top events impacted • Fault trees impacted • Time to SG overfill impacted (inducing an impact on HRA of OD# top events)	Impact, but only quantitative • Initiator frequency impacted	Impact, but only quantitative • Initiator frequency impacted	Impact • New seismic fragility evaluation • Fault trees impacted

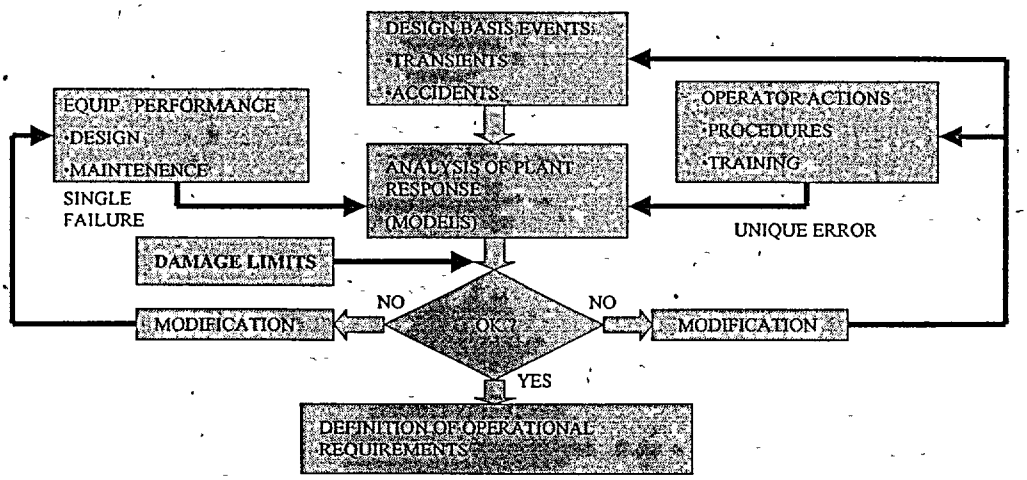
NATIONAL ACTIVITIES IN THE AREA OF SAFETY MARGINS MANAGEMENT

Impact of plant up-rate on PSA level 2 - example of some phenomena categories:

Phenomenon	Impact
1. Prior vessel failure	1. Effect on the timing only. No impact expected on the results. New evaluation of the in-vessel hydrogen production required.
2. Blowdown and pressure spike at vessel failure	2. No impact
3 Pressure spike due to DCH	3. No impact
4 Pressure increasing due to quench	4. No impact
5 Long term overpressure leading to containment failure	5 Reduction in the time to containment failure, but bounded by the definition of the four time frames No impact.
• Probability to quench the debris	• Small increasing in the conditional probability of no debris quench, but considered as small. No impact

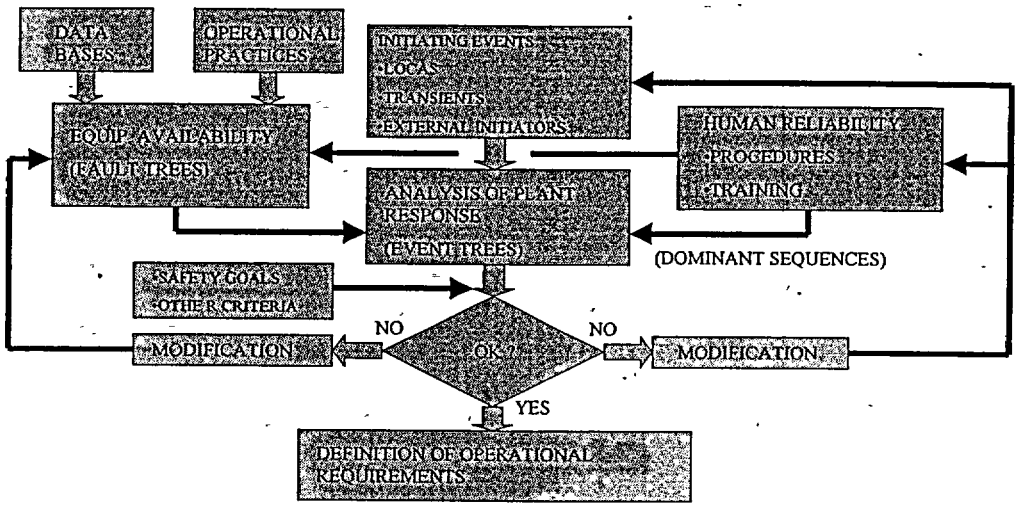
THE DETERMINISTIC APPROACH

SAFETY ANALYSIS - DETERMINISTIC



THE PROBABILISTIC APPROACH

SAFETY ANALYSIS - PROBABILISTIC



GRS

Activities in Germany

Horst Glaeser

Meeting on CSNI Action Plan in the Area of Safety Margins
21-22 November 2002
Paris, France

German Practice

- 2 kinds of conservative assumptions

  a) limited availability of components and systems
  ⇒ single failure and unavailability due to preventive maintenance

  b) assumptions to consider insufficient knowledge by conservative initial and boundary conditions

- RSK Guidelines allow latitude towards application of best estimate models and assumptions
- Rules and guidelines allow to use reliable results from research and development
- No request in rules and guidelines to apply an evaluation code with frozen conservative assumptions

?

German RSK/ French GPR recommendation for EPR safety demonstration

- Compliance of results with licensing criteria has to be proven at high confidence level by

  A) Evaluation of associated uncertainties

  B) Conservative approach

Status of "best estimate" methods in German licensing regulation

- Use of "best estimate" codes possible in the past already, in combination with conservative initial and boundary conditions
- Efforts are underway to include realistic analyses and uncertainty evaluation in the licensing regulation
  - Revision of KTA-Standards (Nuclear Standards Committee) in progress considering best estimate analyses
  - Consideration in licensing activity of the Ministry of Environment and Reactor Safety
  - Topic for Reactor Safety Commission

Various applications by utilities

- Increase of power
- Increase of burnup
- Increase of enrichment

Uncertainty evaluation

- 2 methods developed and applied by
  - Framatome ANP (former Siemens/ KWU)
  - GRS

Planned future uncertainty analyses

- Thermal-hydraulics in reactor coolant system,
  ATHLET computer code calculations
  - PWR, 2 x 100% cold leg break, design basis accident (under way)
  - Basis for fuel failure rate analysis
- Containment,
  COCOSYS computer code calculations
  - Experiment HDR T31.5; large break of main coolant pipe
  - Steam and gas release into containment according to low pressure scenario of German Risk Study
  - Short term phase (pressure decrease in primary coolant system):
    Maximum containment pressure and temperature;
    dependence on input parameters
  - Long term phase (20 h):
    Injection of steam and helium - hydrogen - mixture;
    hydrogen distribution and dependence on input parameters