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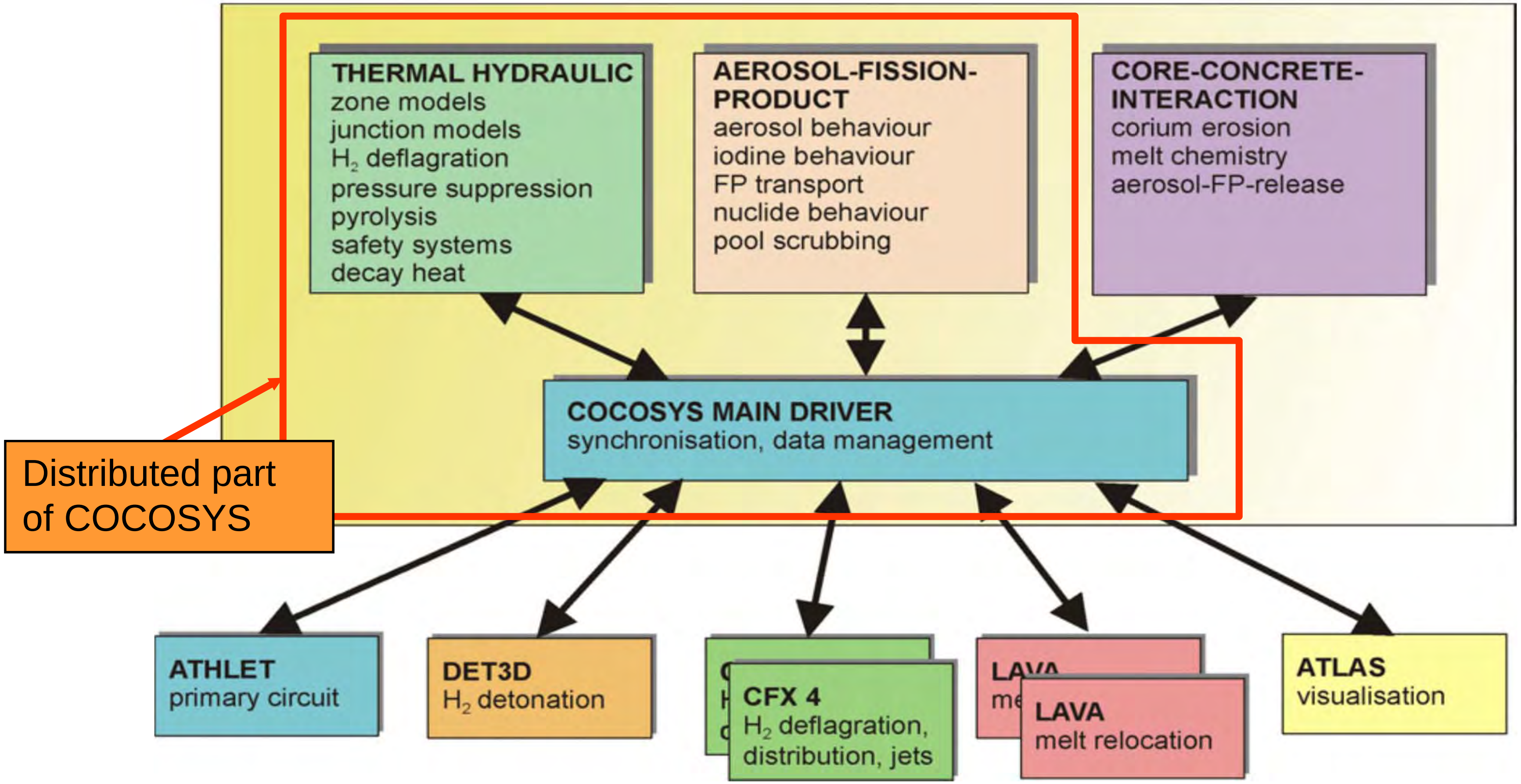
COCOSYS - New Modelling of Safety Relevant Phenomena and Components

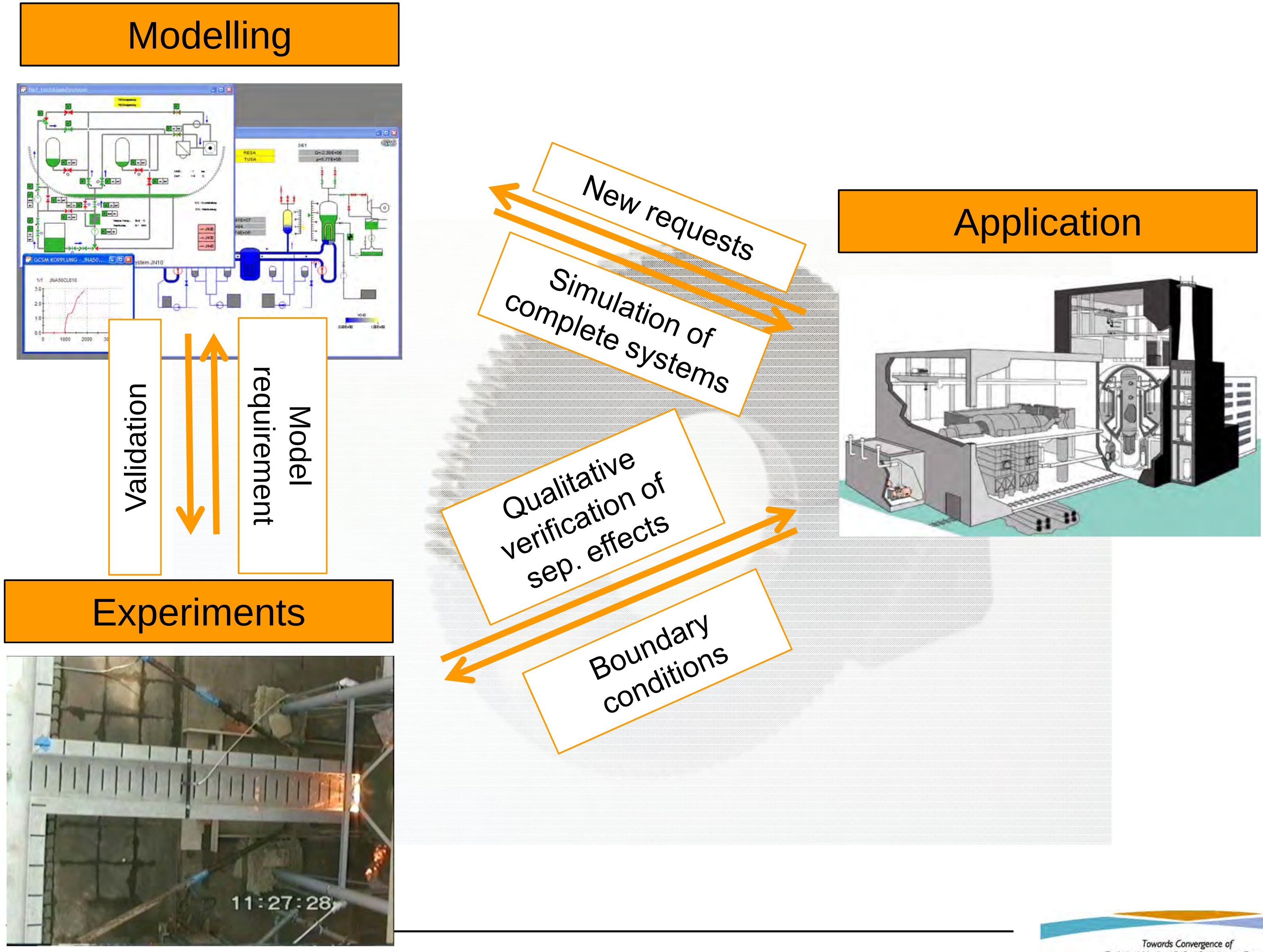
Overview of COCOSYS

- Objective of Containment Code System
  - Provision of a code system on the basis of mechanistic models for the best estimate and comprehensive simulation of all relevant processes and plant states during severe accidents in the containment of light water reactors
  - Also covering the design basis accidents
- Relation to ASTEC
  - CPA (Containment Part of ASTEC) is based on COCOSYS
  - COCOSYS is a reference code for ASTEC
- COCOSYS and ASTEC are already widely developed covering a large number of severe accident phenomena

Structure of COCOSYS System



Link between development, validation and application



Overview of activities

- New requests for applications
  - THY: film evaporation , water flow along structures (*AP 1000*)
  - AFP: Simplified simulation of aerosol and FP release by flashing
- Extented Modelling
  - CCI:
    - Replacement of WEX by MEDICIS (from ASTEC)
    - New options of heat transfer modelling

      ► *Better simulation of 2D MCCI experiments*

Overview of activities (cont.)

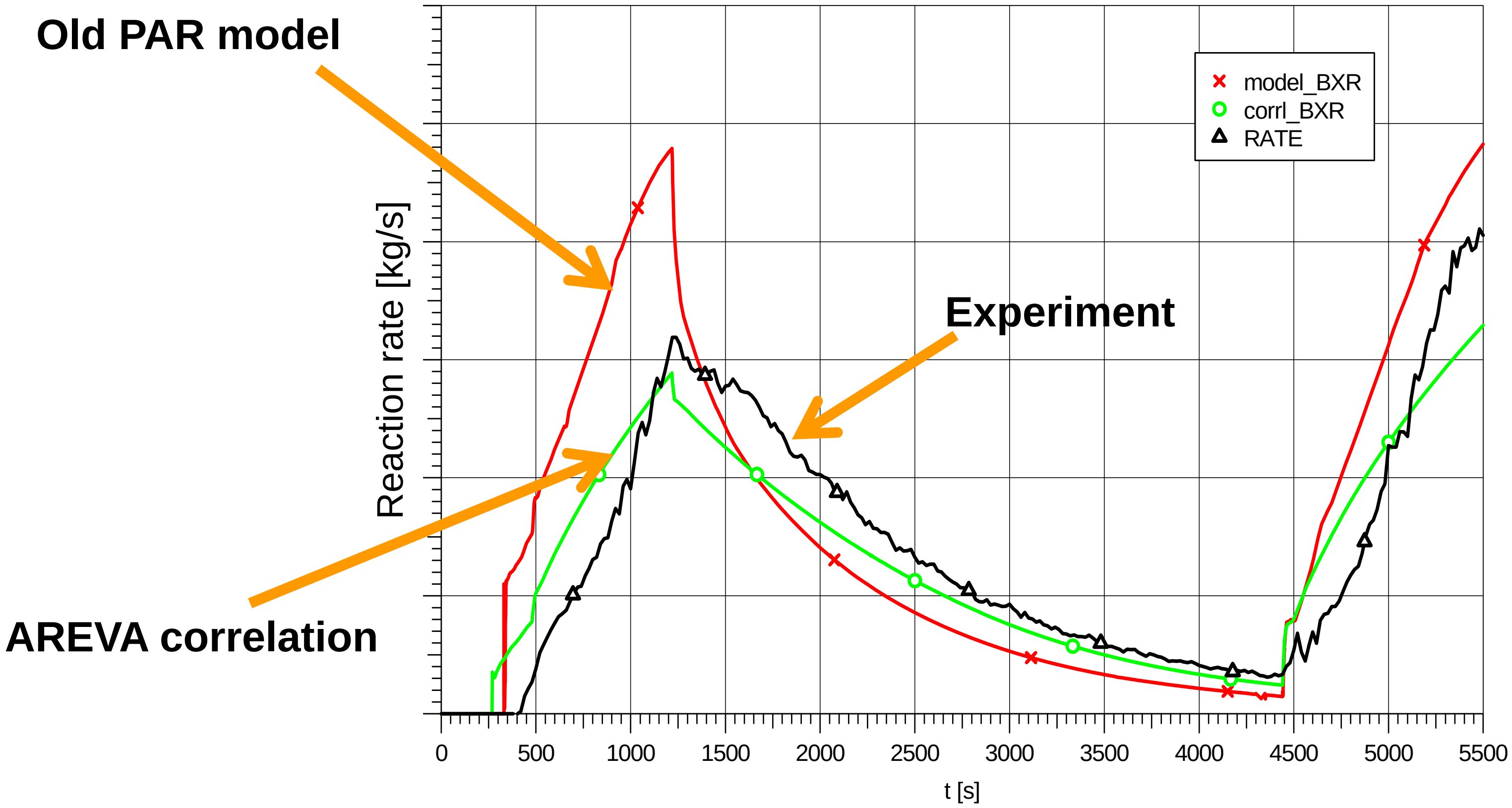
- New experimental results / additional phenomena
  - THY:
    - Development of nodalisation concepts for LP code COCOSYS
    - Improvement of catalytic recombiner model of AREVA type
    - Pyrolysis modelling
  - AFP:
    - Dry and wet resuspension models
    - Iodine chemistry

Topic 1: Improved catalytic plate type recombiner model

- Introduction
  - Previous model uses a **kinetic Arrhenius type reaction model**
  - Recent THAI HR experiments are not well calculated by the old model
  - Experimental studies at Forschungszentrum Jülich demonstrate:
    ► reaction rate is limited by diffusion of reactants to the catalytic plates and **not** by reaction kinetics
  - Implementation of a new **diffusion controlled PAR model**

Topic 1: Improved catalytic plate type recombiner model

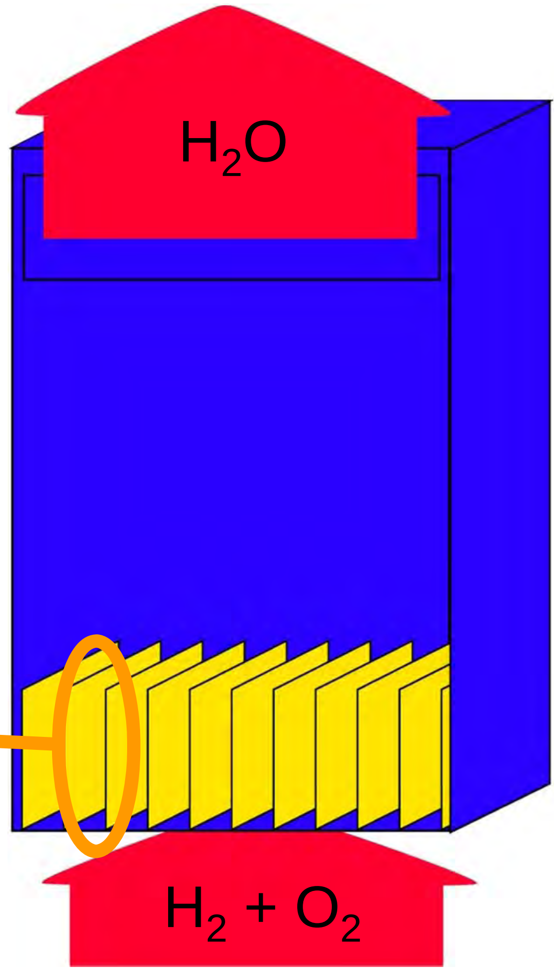
- Introduction (cont.): THAI HR-2 Application of existing model



Topic 1: Improved catalytic plate type recombiner model

- Theory
  - Similar concept of 1D junction
  - Subdivision of plate into 100 parts
  - Mass transfer coefficient β is a function of the position along the catalytic plate and of the diffusion coefficient *D*
  - Successive calculation of *C* as a function of plate height

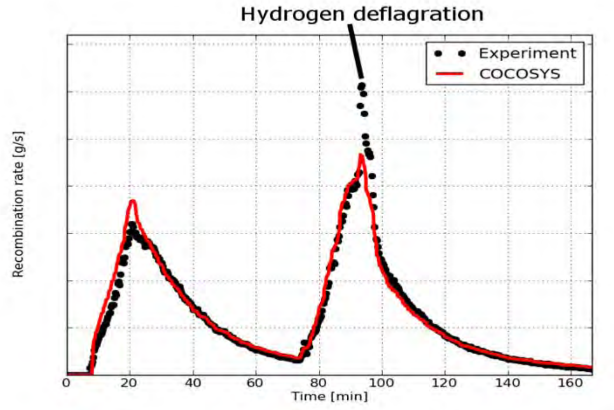
$$C_n = C_{n-1}\frac{v_{n-1}}{v_n} - \beta_n C_{n-1}\frac{A_{kat}}{A}$$



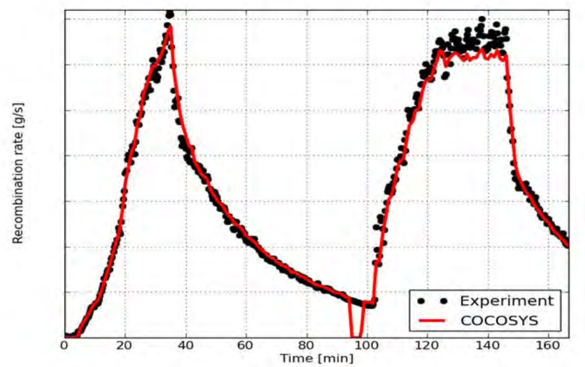
Topic 1: Improved catalytic plate type recombiner model

- **Experiments at 1 & 3.3 bar**
- No steam
- Good agreement with experimental recombination rate
- Slight overestimation in the initial phase

THAI HR-2 (1 bar/dry)



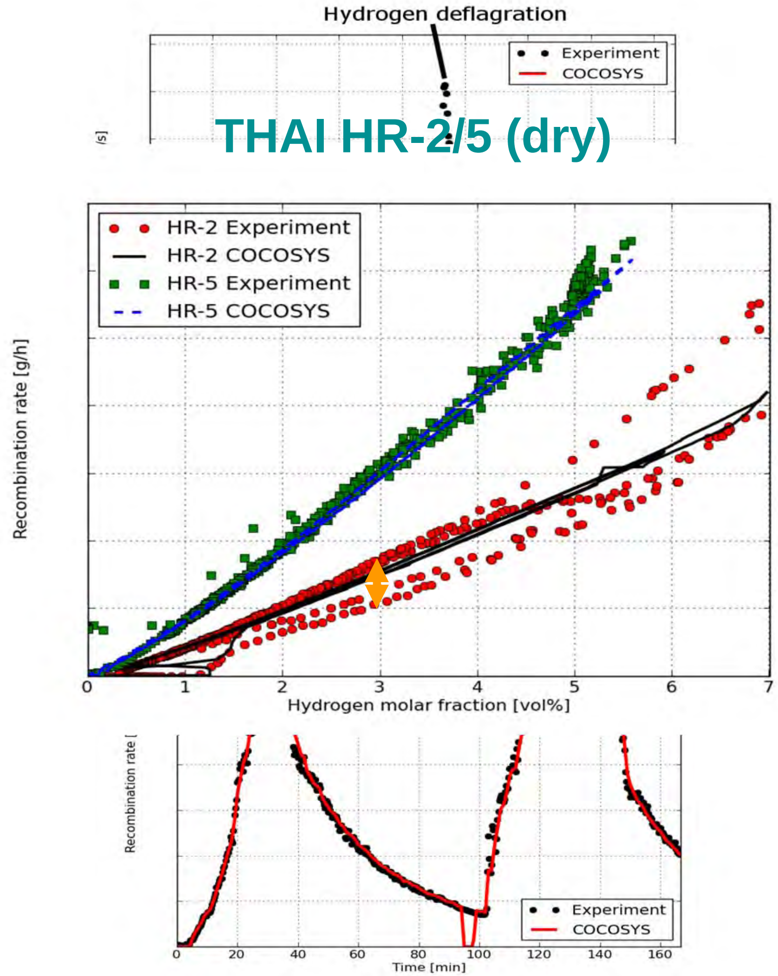
THAI HR-5 (3.3 bar/dry)



Topic 1: Improved catalytic plate type recombiner model

- **Experiments at 1 & 3.3 bar**
- No steam
- Good agreement with experimental recombination rate
- Slight overestimation in the initial phase

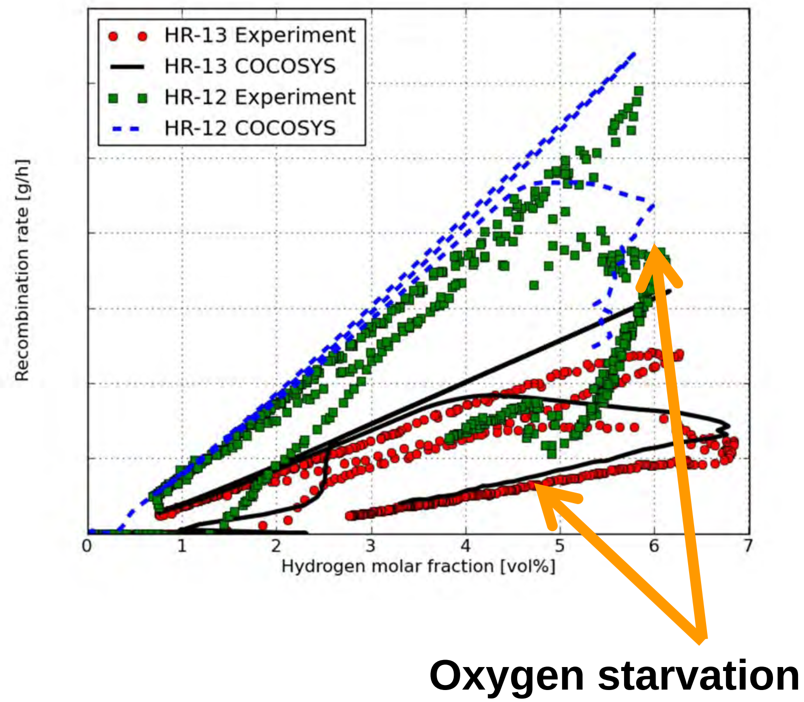
THAI HR-2 (1 bar/dry)



Topic 1: Improved catalytic plate type recombiner model

- **Experiments at 3 & 1 bar**
- 60 vol-% steam
- Good agreement with experimental recombination rate
- Steam slightly reduces the reaction rate → not calculated by diffusion model
- Oxygen starvation considered

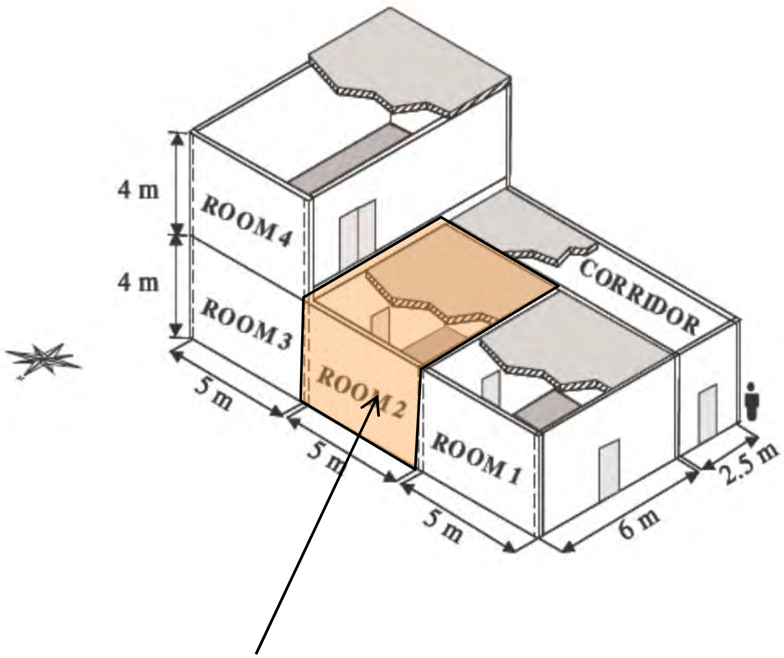
THAI HR-12/13 (60%)



Topic 2: Validation of improved pyroylsis model on PRISME SI-D1 experiment

- Objectives:
  - Heat and soot propagation
  - Effect of ventilation systems
  - Thermal load on targets (cables)
- Fire compartment volume: 6 x 5 x 4 m^3
- Pan size: 0.4 m^2
- Fuel: 5 cm Hydrogenated Tetra-Propylene (TPH, $C_{12}H_{26}$)
- Ventilation: air exchange rate of 4.7 1/h

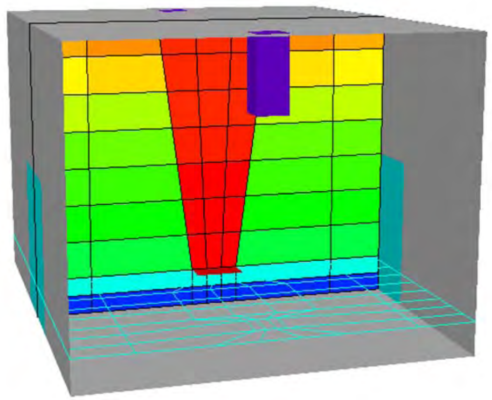
DIVA facility (IRSN)

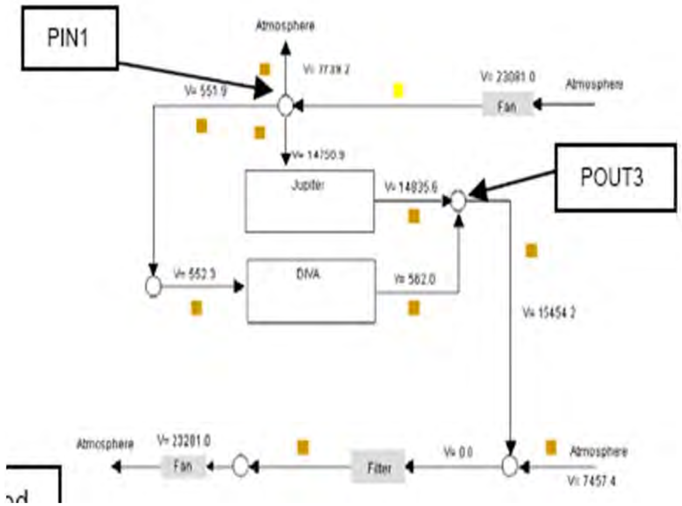


Fire compartment

Topic 2: Validation of improved pyroylsis model on PRISME SI-D1 experiment

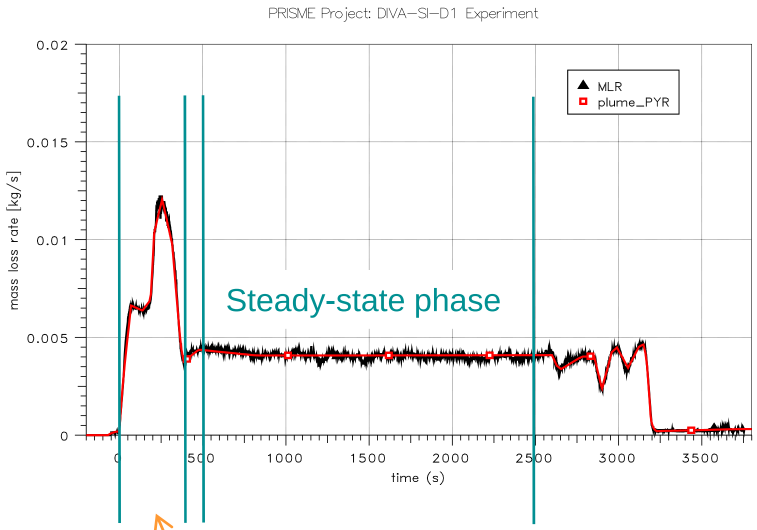
- Nodalisation
  - 297 control volumes (10 levels)
- Models
  - Use of experimental mass loss rate (MLR)
  - Use of AFP to simulate soot
  - Two step reaction:
    - $C_nH_mO_p + ½ [n\text{-}p+ ½m]O_2 \rightarrow nCO + ½mH_2O$
    - $CO + ½O_2 \rightarrow CO_2$
- Simulation of ventilation system





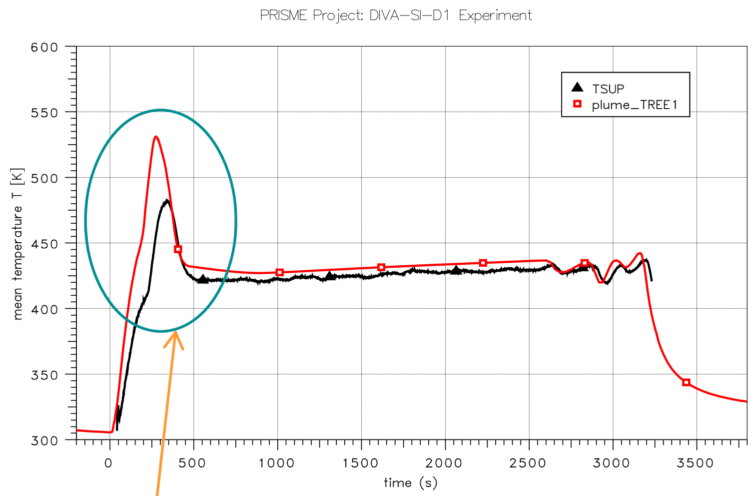
Topic 2: Validation of improved pyroylsis model on PRISME SI-D1 experiment

- Mass loss rate (MLR)
- Mean room temperature



Use of exp. data

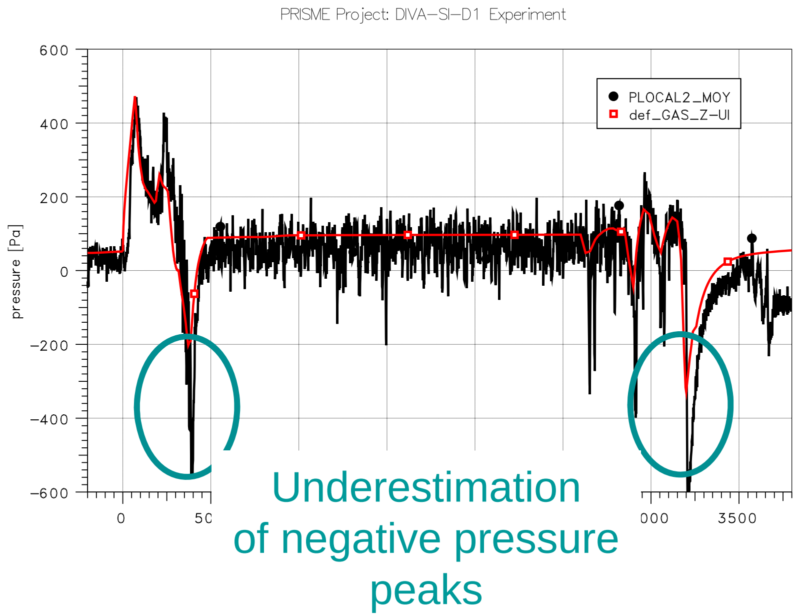
Initial phase

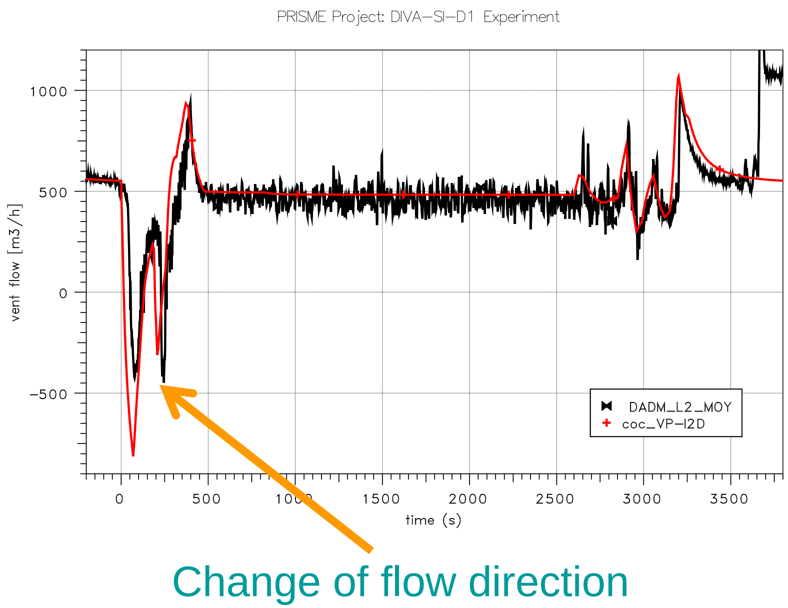


Over-estimation of temperatures during initial phase

Topic 2: Validation of improved pyroylsis model on PRISME SI-D1 experiment

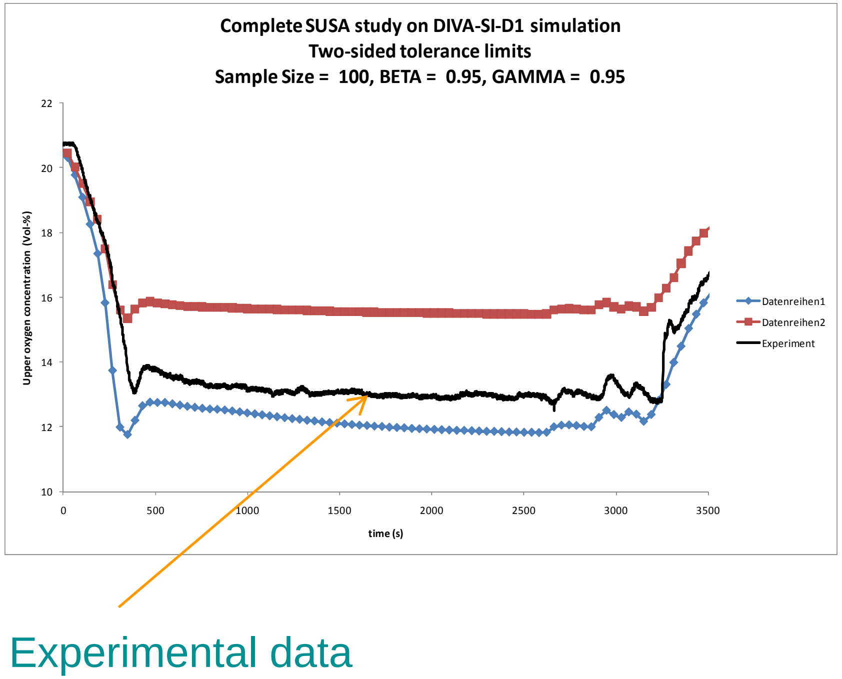
- Pressure inside fire room
- Inlet volume flow rate





Topic 2: Validation of improved pyroylsis model on PRISME SI-D1 experiment

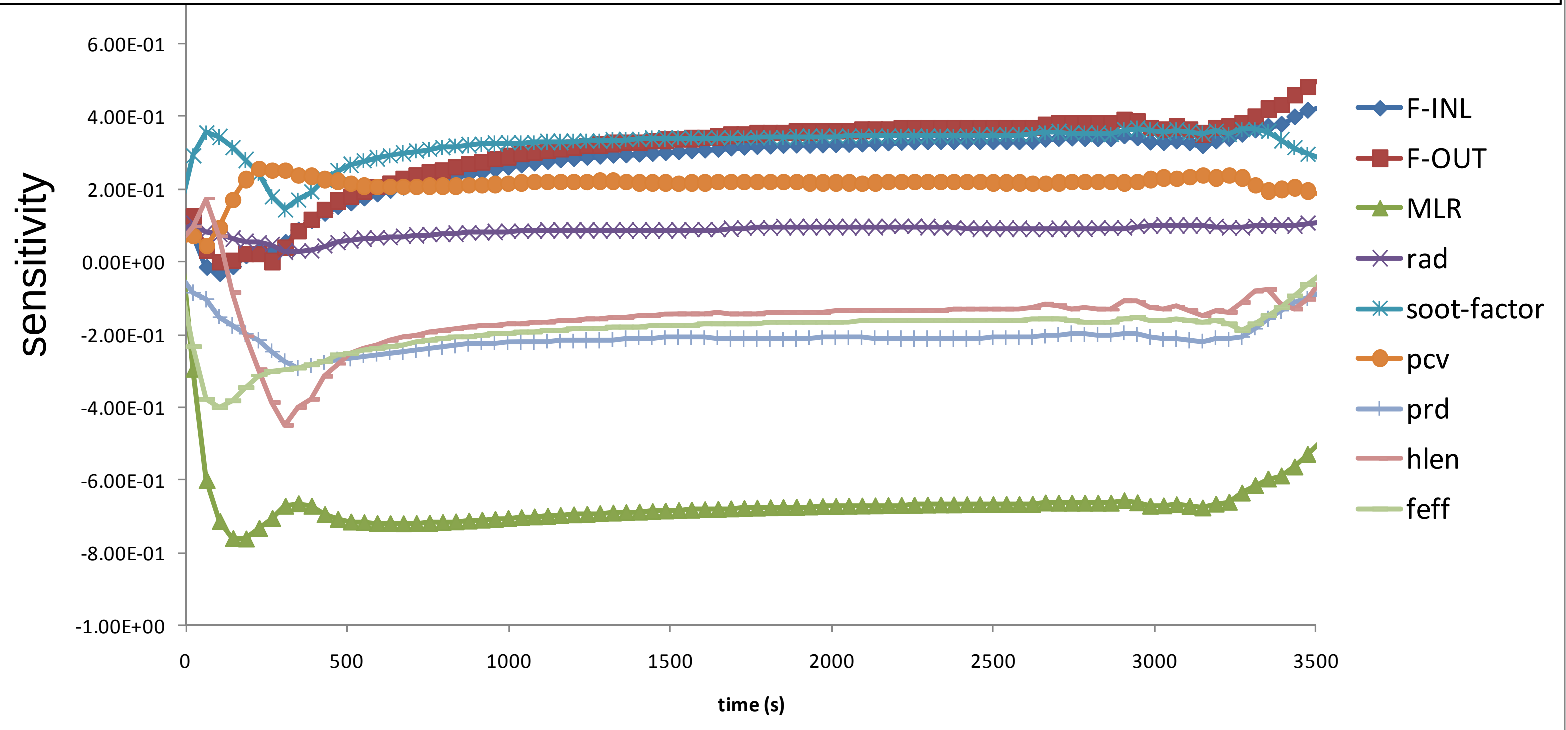
- Uncertainty and sensitivity study with GRS tool SUSA
  - Selection of 62 uncertain parameters
    - Experimental boundary conditions
    - Material properties
    - Aerosol type input variables
    - COCOSYS specific input variables
- Exemplary result: Upper oxygen concentration



Topic 2: Validation of improved pyroylsis model on PRISME SI-D1 experiment

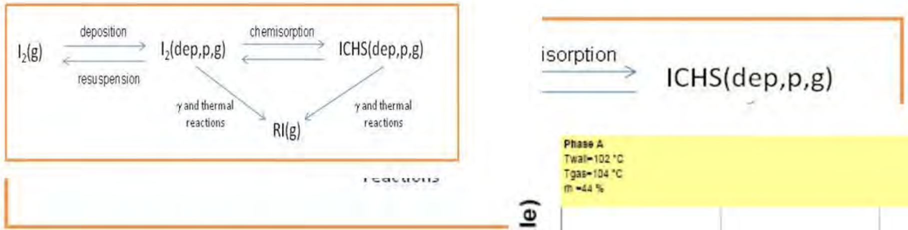
Most sensitive parameters:

- ▶ all time: mass loss rate (MLR), soot factor (f_s)
- ▶ initial phase: convection (pcv, h_{len}), efficiency (f_{eff})
- ▶ steady state phase: ventilation (F-INL, F-OUT), radiation (prd)

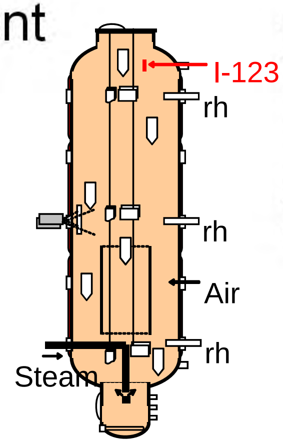


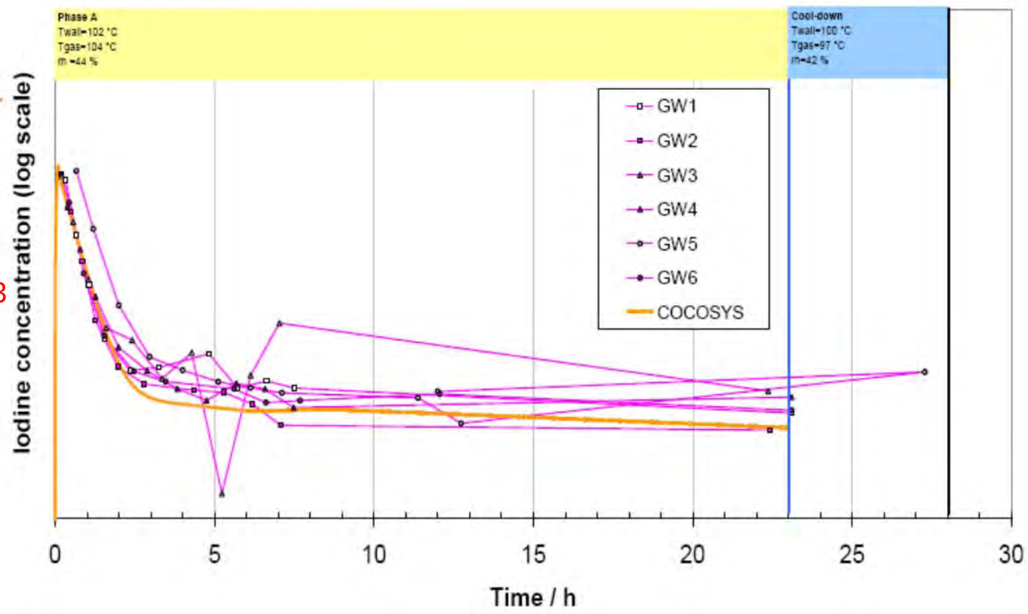
Topic 3: Improved Iodine Chemistry Module AIM-3

- Iodine interaction with painted walls
  - Addition of a "chemisorbed" iodine species



  - THAI-IOD 15 experiment





Topic 3: Improved Iodine Chemistry Module AIM-3

- I_2/Ozone Reaction
  - I_2 (g) is oxidized by air radiolysis products, e.g. ozone (O_3), into **iodine oxides (IO_x)**

$$3\ O_2 + \gamma \leftrightarrow 2\ O_3$$
$$I_2(g) + O_3 \leftrightarrow 2\ IO_x$$

- Physical behaviour changed:

  reactive gas (I_2) → fine disperse, soluble aerosol (IO_x)

  ► IO_x is simulated as a separate aerosol component

- IO_x aerosol measured in **THAI** and evidence in **Phébus** tests

Conclusions

- Topic 1: Recombiner (AREVA type)
  - New diffusion type model simulates oxygen starvation effects
  - But still some overestimation of reaction rate, no simulation of ignition
- Topic 2: Fire simulation
  - Simulation of feed back of fire on ventilation is possible
  - Uncertainty and sensitivity studies should become a common praxis
- Topic 3: Iodine chemistry
  - New models based on THAI iodine experiments
  - Suspended IO_x is simulated as a separate aerosol component

Outlook

- New AFP structure
  - due to new requirements on AFP simulation
    - Local dose rate; improved wash-down simulation
    - I_2 - aerosol interaction (THAI experiments)
- Ongoing uncertainty and sensitivity study on iodine model
- Completion of ex-vessel simulation, melt-relocation, DCH
- Continuing participation on international networks (SARNET, OECD projects)

Jean Gassino - Pascal Régnier

Assessment of the overall Instrumentation & Control architecture of the EPR FA3 project

Instrumentation and Control

- Level 0: process interface
  - Sensors (temperature, pressure, neutron flux, speed, …)
  - Actuators (pumps, valves, rod control, …)

- Level 1: PLCs (industrial computers)
  - Controllers (closed-loop, open loop)
  - Protections
  - Monitoring

- Level 2: control room
  - HMI (screens, keyboards, mouse)
  - Workstations, mass-storage
  - Hardwired safety panel

I&C main characteristics

- Participates to almost all lines of defence
- Different safety classes

  e.g. Protection System > Post accidental > HMI > non classified
- Protection, control, monitoring
- Automatic and manual functions
- Thousands of physical parameters
- Many sensors/actuators shared by different SC
- Requirements for correctness/safety but also for advanced HMI

30 years of digital I&C in French NPPs

- 1980's: digital Protection System, traditional architecture
  - Acquisition and processing units: SPIN technology
  - Point-to-point links, 8 bit microprocessors, assembly language
  - 2oo4 voters for ESFAS still hardwired
- 1990's: digital architecture
  - Protection system (including voters), NIS, Rod position: SPINLINE technology
  - Networks, 16/32 bit microprocessors, C language, generic system software
  - Computerized control room (with hardwired backup panel)
- EPR: integrated digital platforms
  - Highest Class : mainly Teleperm XS
  - Other Classes : mainly SPPA-T2000 (HMI + PLCs)

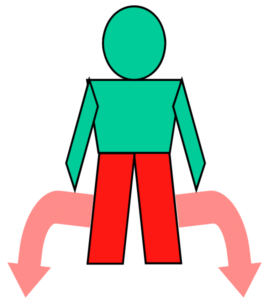
Today

- Almost all systems are computerized
- 24 reactors with computerized Protection System, no hardwired backup
- Increasing interconnectivity between systems

IRSN assessment: based on 2 legs

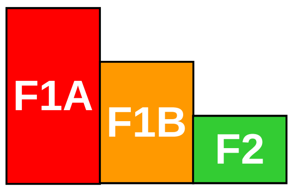
- IRSN has been reviewing digital I&C since the early 1980s

Safe digital systems — #1



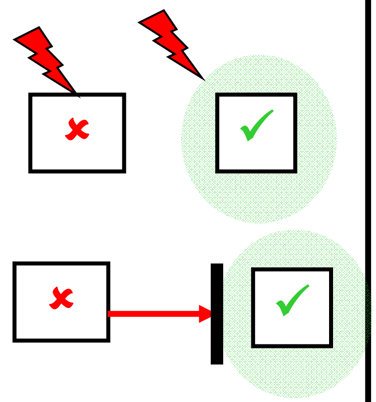
#2 — **Robust architecture**

•Strict development cycle
•Correct specification of requirements
•Sound design
•Relevant verification strategy



Tolerate postulated CCFs

Stop failure propagation



1rst leg: assessing the safety of each digital system

- Software (and programmed logic, e.g. FPGAs) ≠ hardwired logic
  - Development errors vs. random faults due to wear and tear
  - Present since the beginning (or will never happen) vs. appear randomly in operation
  - Infinite set of potential errors (actual ones are unknown) vs. few failure modes
  - ⇒ HW analysis approach not applicable to SW: no FMEA, no quantified reliability…

- Safety software ≠ general purpose software
  - General: many functions, loose life cycle, loose design principles -> <u>quick release</u>
  - Safety: dedicated functions, strict development, strong design principles -> <u>verifiability</u>
  - ⇒ Software may be very reliable provided a dedicated approach is strictly enforced

-> IRSN performs a detailed technical assessment

- analyses of design and verification documentation, including source code

For details about 1rst leg, please see paper (same authors) at NPIC&HMIT 2010

2nd leg: assessing the robustness of the architecture

- Issues: propagation of failures, CCF

-> Postulate failures of systems or other adverse events

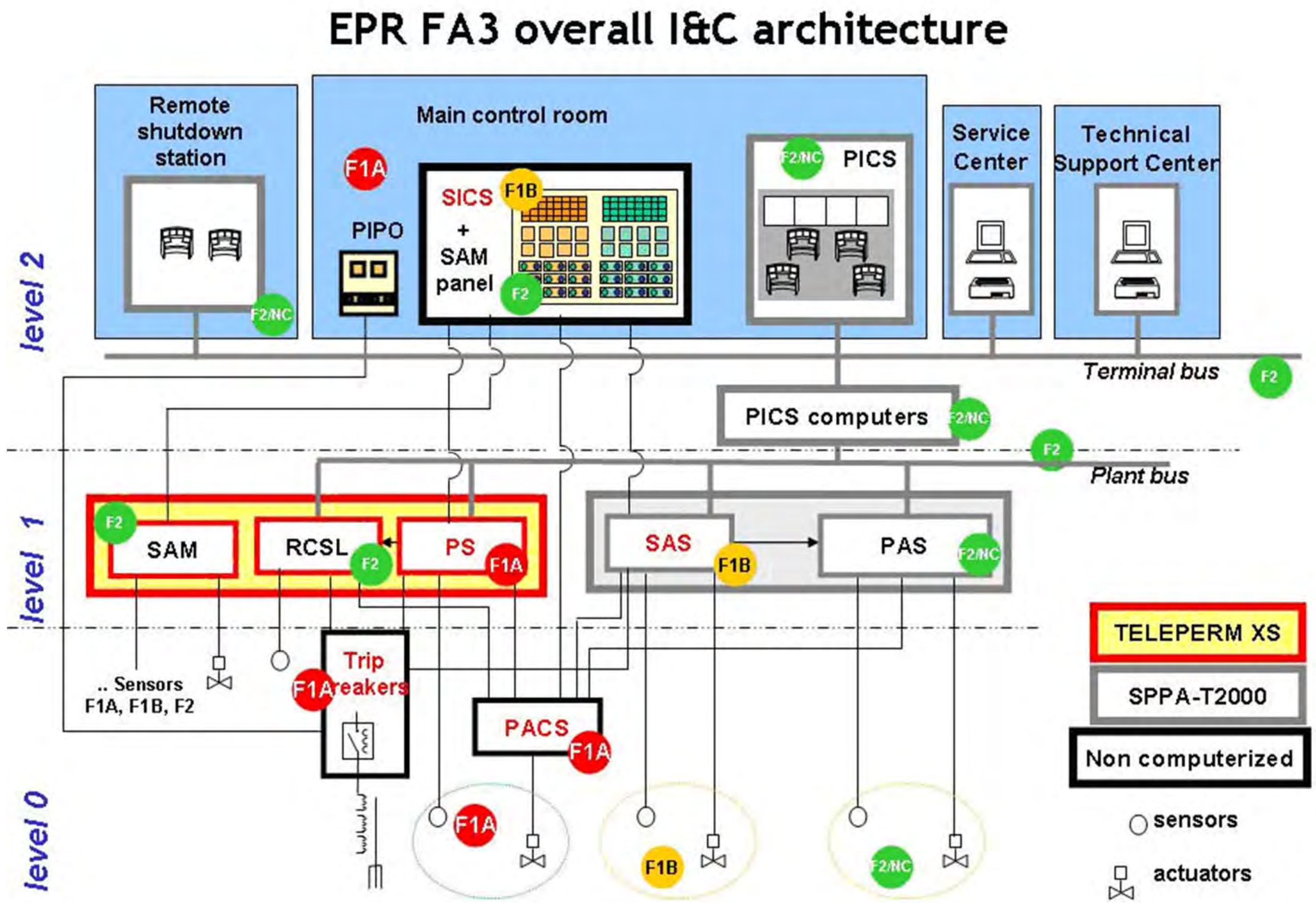
-> Study the impact on other systems or on multiple systems

-> Check whether the safety functions are still available

Identifying the architecture

- I&C systems: roles, functions, boundaries, interfaces, technologies, …
- Defense lines: definitions, roles, systems involved in each line
- Classification: definition of safety classes, associated requirements, class of each system, …

Architecture diagram
(based on public Preliminary Safety Report)



I&C systems

- PS: Protection System
- SAS: Safety Actuation System
- PACS: Priority Actuation and Control System
- SAM: Severe Accident Management system
- RCSL: Reactor Control Surveillance and Limitation
- PAS: Process Actuation System

- PICS: Process Information and Control System
- SICS: Safety Information and Control System
- PIPO: inter operator console panel

Some characteristics of EPR I&C (1)

- 3 safety classes: F1A, F1B, F2 (+ NC)
- 3 lines of defense involving I&C

  - L I: prevention of incidents & accidents (includes limitations)
  - L II: prevention of core/fuel meltdown (PCC2 to 4 and RRC-A situations)
  - L III: prevention of major and early releases (severe accidents = RRC-B)

- 2 platforms (families of equipment and related engineering tools)

  - Teleperm XS from AREVA

    mainly for Protection System (F1A) and Limitations
  - SPPA-T2000 from Siemens

    for post-accidental (F1B), HMI (F2), other F2 and NC

Some characteristics of EPR I&C (2)

- FA3: non digital priority management of actuators (PACS)
  - PACS manages orders from all defense lines and all SC
- Extensive use of networks (Terminal Bus, Plant Bus)
- Computerized HMI (F2)
  - Built on general purpose Operating Systems
  - Used in all plant situations as long as available
  - Backed up by a non-computerized HMI

Architecture, ideal case

- Full independence between any two lines of defense
- Full independence between any two safety classes

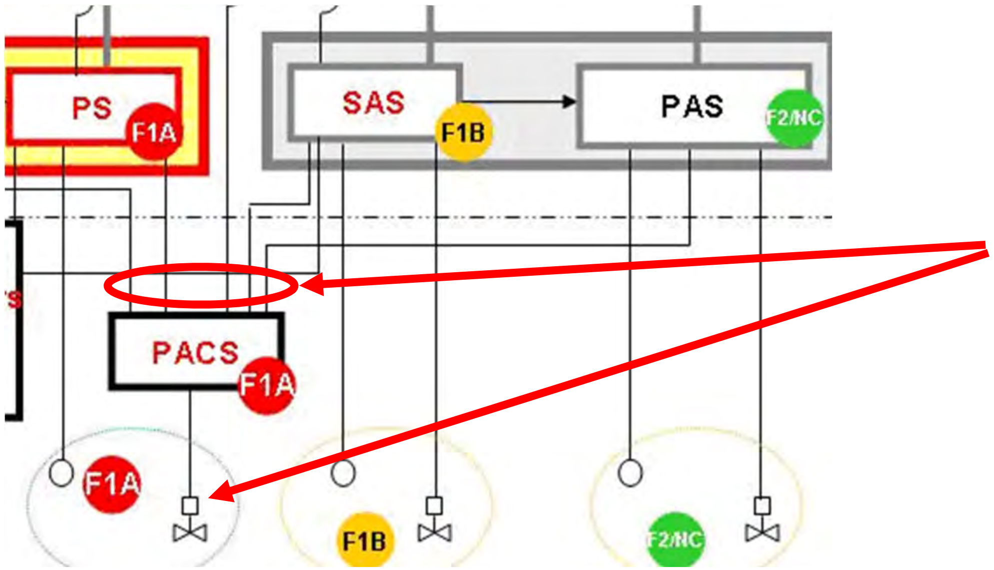
-> Obviously good high level safety principles

-> Difficult to achieve in practice

  – General plant design -> increased number of defense lines
  – Finer breakdown in the safety classification
  – Other technical constraints (next slides)

-> Assessment of independence more difficult than on the first reactor series (only the Protection System had to be independent from the rest of the I&C)

Architecture, constraints (1)



Some actuators are controlled by several Safety Classes or lines of defense

- E.g. a valve operated during normal operation for regulation purposes, must be closed when an accident occurs
- This implies communication from less to higher classified
- Such dependencies, coming from the plant design, are necessarily reflected in the I&C solution

Architecture, constraints (2)

- Sensors of the primary circuit
- Should they be used by only one line of defense ? (e.g. not by protection system and normal controls)
  - Straightforward assessment
  - But need for many additional sensors

- That could weaken the primary reactor coolant system
- The choice is made during plant design

  -> another constraint on the I&C

Architecture, constraints (3)

- Some systems have sometimes to be connected to maintenance terminals
- Should the maintenance terminal have the same safety class?
  - Straightforward assessment
  - But less diagnostic capabilities (highest SC would have the lower diagnostic capabilities)
- Balance to be found

Architecture, constraints (4)

- The same operator will cope with normal situations as well as with incidents and accidents
- Should the same operating mean (i.e. computerized HMI) be used in both situations ?
  - Good for Human Factor aspects
- But this common HMI introduces dependencies in the I&C between different:
  - Lines of defense
  - Safety classes

Architecture, constraints (5)

- An internal hazard like fire or flooding will impact all I&C equipment in the affected zone
- Thus full independence would prevent putting in the same fire zone I&C systems belonging to different safety classes or different lines of defense
- This would imply
  - 4 divisions for the PS
  - Plus divisions for standard I&C, separate from those of PS
  - Plus separate divisions for severe accident management
- This is beyond any current practice

Impact on the assessment

- Full independence between lines of defense and between safety classes
  - is not achievable in practice
  - would not necessarily lead to the best overall plant design
- The assessment is therefore more difficult than it seemed
- Need to accept some exceptions duly justified in terms of safety

Robustness as a new starting point

- Need to come back to the basic safety objectives

  -> ensure that no plausible event affecting a set of I&C devices can jeopardize the availability of I&C functions

- What are the plausible events to consider ?

  (i.e. to what should the architecture be robust?)

Categories of plausible events (1)

IRSN has established the following categorization of plausible events, based on the experience of 3 reactor generations

- Single failures
  - Random failures due to physical degradation; all possible failure modes, not only "graceful failures"
- Internal hazards which can be confined
  - Typically fire or flooding; all possible failure modes have to be considered
- Earthquake
  - Unless proven otherwise; all possible failure modes have to be considered
- Common cause failure by electrical propagation through power line
  - Propagation of electrical faults via the power supplies of I&C
- Common cause failure by electrical propagation via the signals
  - Propagation of electrical faults via electrical link within I&C

Categories of plausible events (2)

- Technological common cause failure (CCF)
  - Simultaneous failure of two or more items of the same model or range, the root cause being a design error

  -> This does not extend a priori to a technical domain in general (if adequate diversity is demonstrated)
  E.g. failure of all software based equipment, on the sole basis that it includes software, is not postulated

- CCF by propagation of erroneous information or by dependency on common implicit information
  - One device transmitting erroneous data to other devices
    - Regardless of the transmission technology, even if is more sensitive for networks (which tend to increase the amount of transmitted data)
  - Dependency on implicit information
  e.g. problems with range limitations, switching from summer to winter time

Tolerance level

Represents the location of the countermeasure to a given event

- Equipment
  - The piece of equipment can tolerate the event (e.g. seismic qualification)
- System
  - The system can tolerate the event (e.g. a redundant system vs. single failure)
- Line of defense
  - Another system or provision can tolerate the event within the line of defense considered (e.g. ATWS)
- Architecture
  - The event can impact several lines of defense, but the overall architecture can still perform the functions (e.g. PICS used for all plant situations and SICS as a backup)
- Exclusion
  - The event is excluded for a given type of equipment (typically the technological CCF is excluded for components having a "simple" design)
  - The exclusion must be justified ("excluding" is not "ignoring")

Screening of plausible events (1)

- IRSN has applied its assessment methodology to each device x event
- And identified the “tolerance level” provided by the architecture

	Level 0	Level 1	level 2
Single failure	1 System		
« Sectorizable » hazards	2 System		
Seism	3 Equipement		
Technological CCF	4 analog: Excluded CEP: overall architecture	6 Line of defence	8 Overall architecture
Propagation of erroneous information	5 Excluded or reported to level 1	7 Overall Architecture	9 Overall architecture
Propagation via power supply	10 Overall architecture		
Propagation via electrical signals	11 System		

Screening of plausible events (2)

- 11 topics (device x event) were then systematically screened
- If EDF's claim was not self evident, a detailed analysis were performed
- E.g. topic 7 (propagation of erroneous information at level 1)
  - IRSN identified issues on the Plant Bus and required justifications/changes
  - Analyses involved details far below the overall I&C diagram
- E.g. topic 4 (technological CCF on smart sensors)
  - IRSN identified some smart sensors and requested EDF to postulate a CCF on any given smart sensor reference
  - EDF had to verify that a given reference was not used for the same accident sequence in different lines of defense (otherwise diversify)

Outcomes of the assessment (1)

- Conclusions and requests presented by IRSN in 2009 to the standing board of experts and followed up by ASN letter
  - Addition of an independent mean to validate HMI commands that could inhibit F1A functions
  - Addition of F1B mean to detect postulated failures of computerized HMI (for reliable detection of when to switch to non-computerized SICS)
  - Demonstration of non perturbation of the HMI (F2) by NC devices (justifications + changes)
  - Backup of SPPA-T2000 F1B functions in a dedicated TXS based system
  - Extension of the SICS coverage to risk reduction categories A and B
  - Clarification of the operating means available in case of failure of the SPPA T2000 platform during a severe accident

Outcomes of the assessment (2)

- In addition during the assessment itself, the technical discussions led EDF to propose some significant improvements
  - E.g. introduction of a dedicated “SAS Bus” for SAS communications (rather than using the Plant Bus shared with F2 and NC)
- During summer 2010, IRSN received as scheduled the technical proposals and justifications from EDF
  - The examination of those answers is in progress and has up to now not led to identifying any new significant architectural issue

Conclusion (1)

- Assessment of the EPR I&C architecture proved quite challenging: high degree of interconnectivity, comprehensive digital platforms
- A method has been devised and applied to assess safety without needing everything being independent of everything
- Reviewing the architecture is only half of the task. It complements the detailed technical assessment of the I&C systems, especially software
- Independence and defense in depth must by no mean be considered as a workaround for avoiding a detailed technical justification of safety systems including software

- The changes introduced by EDF in the architecture improve safety

Conclusion (2)

- Regarding the mid term (beyond FA3), we can note 2 particular issues (among all those raised by the fast evolving technology)
- Choice of general purpose computers for the main HMI
  - Benefits regarding human factors
  - But introduces many links between different SC and defense lines
  - Fortunately, technologies for safety computerized HMIs are emerging
- Trend in the automation industry to propose a versatile architecture with all devices connected to networks
  - This will increasingly conflict with independence requirements
  - Designers of safety I&C should instead identify all communication needs and independence requirements and then implement only the required links

Directorate-General for Energy

EUROPEAN COMMISSION

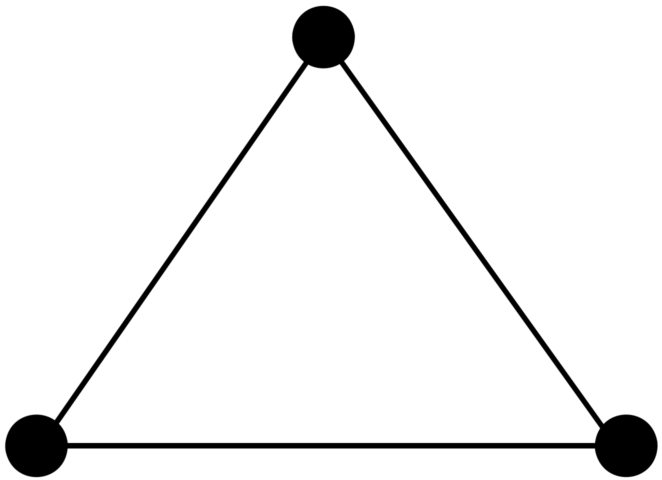
Energy – increasingly a European affair

Ute Blohm-Hieber
Head of unit
Nuclear energy, Transport, Decommissioning & Waste management

EUROSAFE Forum, 8 November 2010

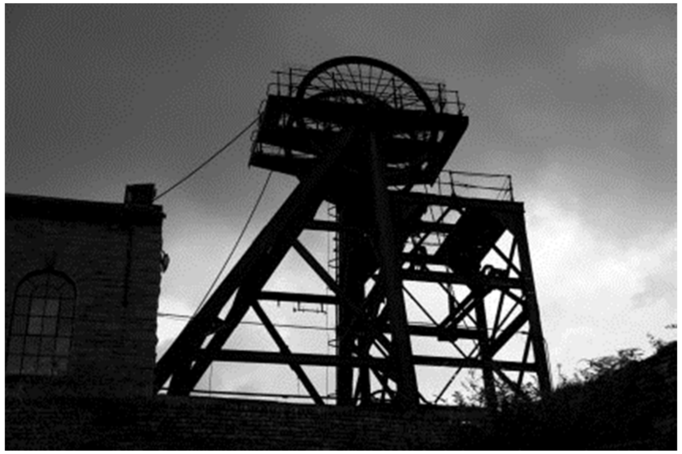
Energy policy in the EU

The challenge for energy policy



Energy as a central idea of the European integration

- European Coal and Steel Community (1951)
- European Atomic Energy Community (Euratom, 1957); reference for further developments within the European Economic Community context



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Energy - Driver of integration

- Integrated EU energy and climate policy (2007)
- Lisbon Treaty (2009) with energy chapter and four related objectives:
  - » ensure the functioning of the energy market;
  - » ensure security of energy supply in the Union;
  - » promote energy efficiency and energy saving and the development of new and renewable forms of energy
  - » promote the interconnection of energy networks.

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Europe 2020 - European strategy for smart, sustainable and inclusive growth

- Seven flagship initiatives, including
  - » Innovation Union
    The aim of this is to re-focus R&D and innovation policy on the challenges facing our society, such as climate change, energy and resource efficiency, health and demographic change
  - » Resource efficient Europe
    The aim is to support the shift towards a resource efficient and low-carbon economy that is efficient in the way it uses all resources
  - » An agenda for new skills and jobs
    The aim is to create conditions for modernising labour markets with a view to raising employment levels and ensuring the sustainability of our social models

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Upcoming Initiatives

- 2011-2020 Energy strategy (11/2010)
- 2050 Energy Roadmap – energy scenarios (2011) including an indicative nuclear programme (PINC)
- Energy efficiency action plan (2011)
- Communication on the external dimension (2011)
- Energy infrastructure package (11/2010)
- Implementation of the SET Plan – Launch of industrial initiatives 15/16 Nov2010

Directorate-General for Energy

EUROPEAN COMMISSION

Nuclear energy at EU level

We all go European in nuclear…

- Technical safety organisations: **ETSON**
- Operational experience: **Clearinghouse**
  - » Participation of TSOs appreciated
- Education and training: **ENELA, ENSTTI, ENEN, EHRO-N, EUSECTRA**

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The role of the European institutions

„The EU has one political priority: the respect of the guarantees of safety, security and non-proliferation.“

“…the most advanced frame and the highest norms…”

José Manuel Barroso
President of the European Commission
Paris, 8. March 2010

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Two basic approaches

Legislation

Other Initiatives

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Nuclear Safety Directive (June 2009)

- Independence of regulators
- Prime responsibility of license holders
- Obligations on transparency
- IAEA SS and CNS provisions legally binding
- EU first major regional actor with binding safety rules

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Waste Directive

- Content
  - » Highest standards for management of radioactive waste and spent fuel
  - » Concrete national programmes
  - » All waste – from generation to disposal
  - » No exports
- Adopted by Commission on 3 November

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Context

- All Member States produce Radioactive Waste (RW)
  - » Nuclear power, medicine, industry and research, spent Fuel (SF) from nuclear reactors
- Existing EU legislation
  - » No specific rules on SF and RW management in the long term
  - » Consistency with the Nuclear Safety Directive
- IAEA Safety Standards and the Joint Convention
  - » Internationally endorsed principles and requirements for SF and RW management

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General principles

- MS to have National policies on SF and RW management
  - » Core: National Programme
- No undue burdens on future generations
- SF and RW to be safely managed in the long term
- Disposal of RW in the MS in which it was generated, "shared" repositories in the EU possible under voluntary agreements between MS

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Obligations

- Competent regulatory authority - independent and given legal powers and resources
- Licence holder - prime responsibility for the safety
- Safety case – to demonstrate that the safety requirements are met; disposal facility – demonstration of passive safety features
- Expertise and skills - education and training
- Funding – ensure adequate financial resources
- Quality assurance of the safety
- Transparency – public information and participation in the decision-making process

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National programmes

- Contents:
  - Inventory;
  - Concepts, plans and technical solutions;
  - Concepts and plans after closure of a disposal facility – institutional control;
  - Description of research, development and demonstration;
  - Milestones, timeframes and responsibilities;
  - Key performance indicators;
  - Cost assessment;
  - Financing scheme(s) to cover all costs

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Notification, Reporting & Review

- Notification of the National Programmes to the Commission
- Regular reports to the Commission
- Peer Review – every 10 years

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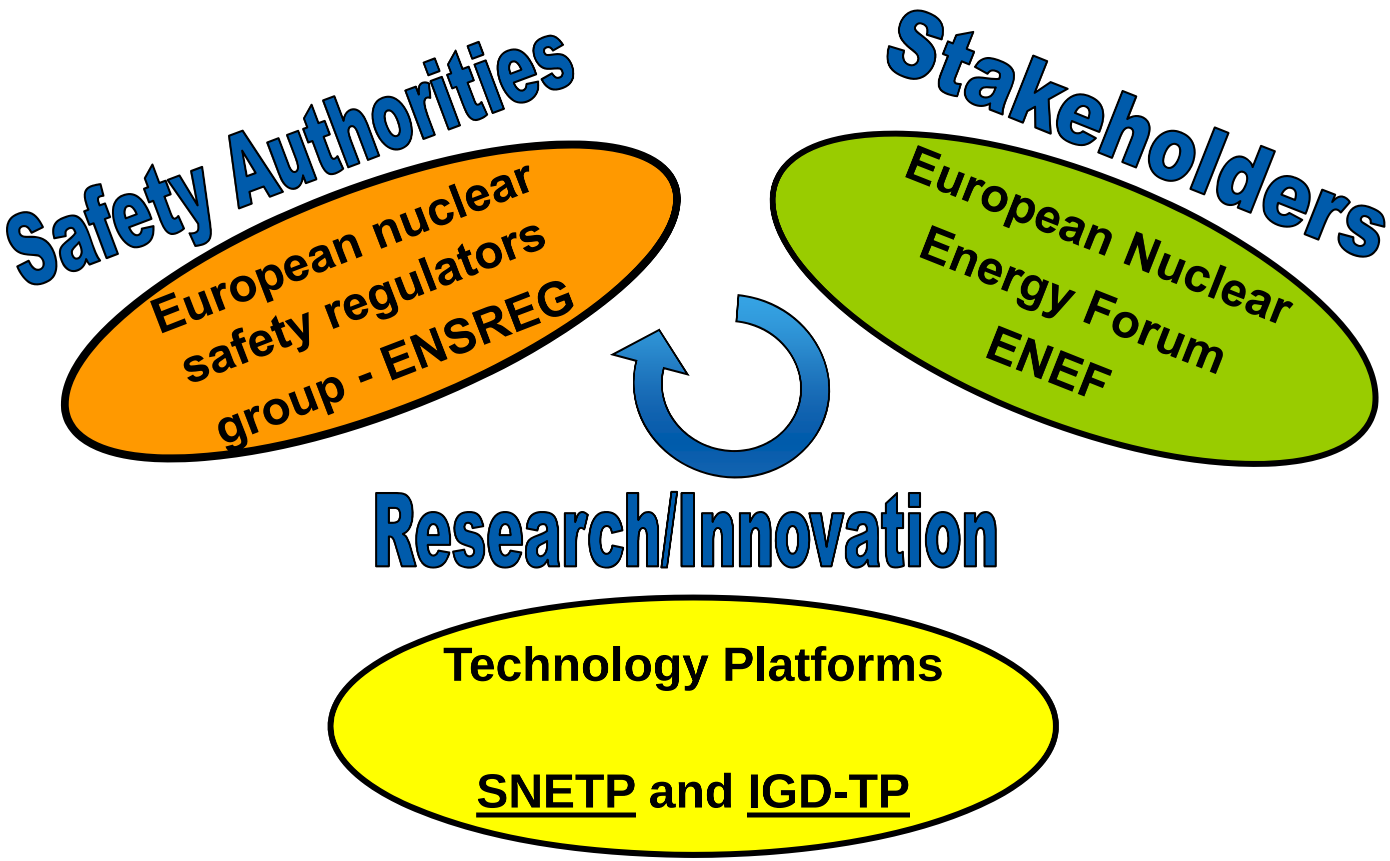
New forms of governance on nuclear



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Conclusion

- Going European in energy policy is necessary and will continue
- EU committed to complete frame for highest standards for safety, security and non-proliferation
- Continuing dialogue with all stakeholders
- ETSON is a key contributor to the dynamic processes at EU level

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If we didn‘t have the European Union, we might need to invent it…

C. Heckötter (GRS) - J. Sievers (GRS) – F. Tarallo (IRSN) – N. Bourasseau (IRSN) - B. Cirée (IRSN) – A. Saarenheimo (VTT) – K. Calonius (VTT) – M. Tuomala (TUT)

COMPARATIVE ANALYSES OF IMPACT TESTS WITH REINFORCED CONCRETE SLABS

OUTLINE

- Introduction
- Impact test facility at VTT
- Hard missile test / simulations
- Soft missile test / simulations
- Summary

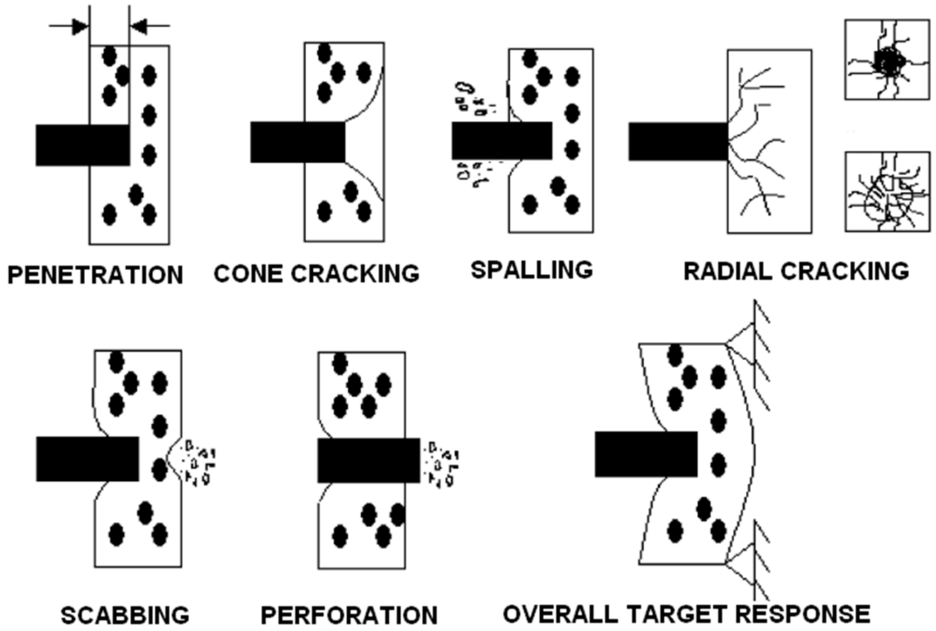
INTRODUCTION

- Integrity of containment structures subjected to impact loadings due to airplane crash is a safety relevant topic for
  - design of new plants
  - long term operation of nuclear power plants
- Objectives of research work on that topic
  - Development
  - Provision
  - Validation

  of structural mechanic analysis methods for the simulation of the phenomena during impact of missiles on structures

IMPACT EFFECTS ON CONCRETE TARGETS

- Local effects
  - Penetration
  - Punching (cone cracking)
  - Spalling
  - Scabbing
  - Perforation
- Global effects
  - Slab deflection
  - Radial cracking



Drawing according to Li, Q.M. et al., "Local impact effects of hard missiles on concrete targets", Int. J. of Impact Eng. 32, 2005

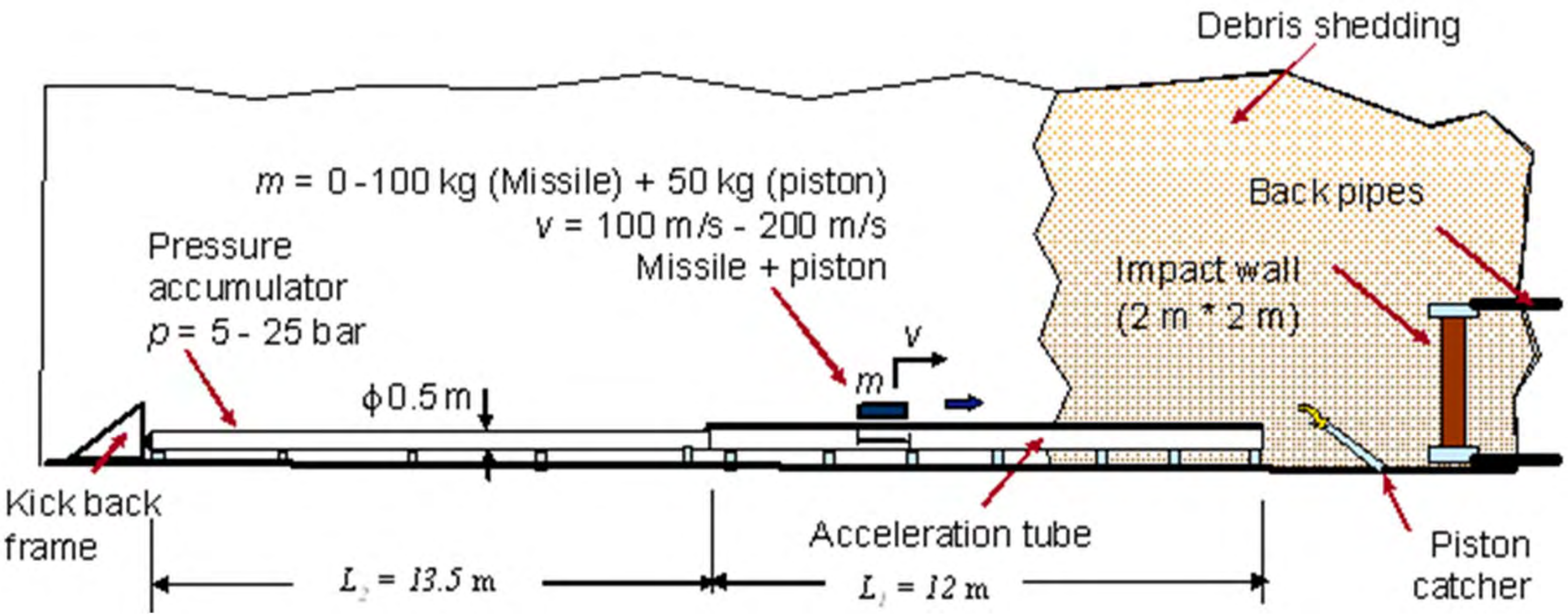
EXPERIMENTAL FACILITIES FOR IMPACT TESTS

- KIT (Germany)
  - Small scale, soft missiles, rigid targets
- VTT (Finland)
  - Intermediate scale, various missiles, various targets
- Meppen (Germany)
  - Large scale, soft missiles, reinforced concrete targets
- SANDIA (USA)
  - Large scale, water tanks, reinforced concrete targets

VTT IMPACT PROJECT

- International program (12 partners)
- Provide data for validation of analysis tools
- Targets
  - Force Plates
  - Reinforced concrete slabs
- Missiles
  - Hard missiles
  - Soft missiles
  - Wet missiles
  - 3D missiles

IMPACT TEST FACILITY AT VTT



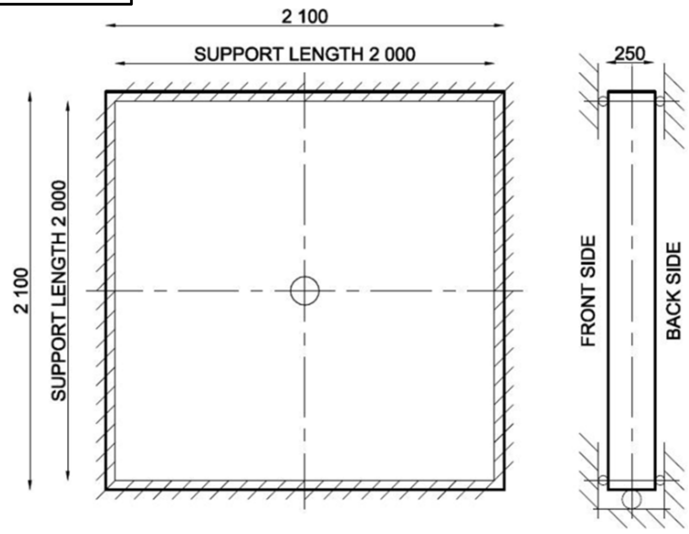
Flexible regarding:

- Target
- Missile

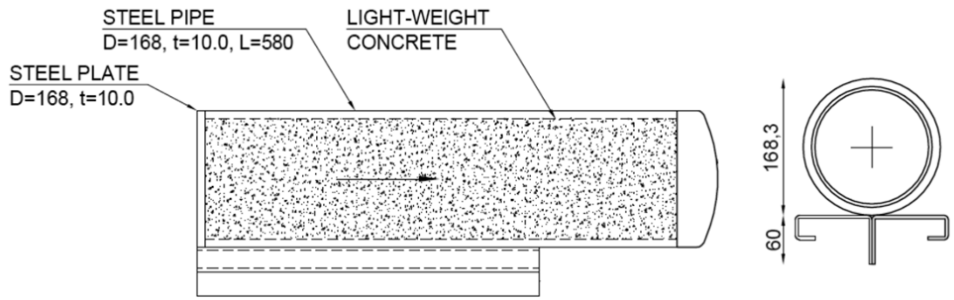
IMPACT TEST FACILITY AT VTT

HARD MISSILE TEST SETUP (TEST 699)

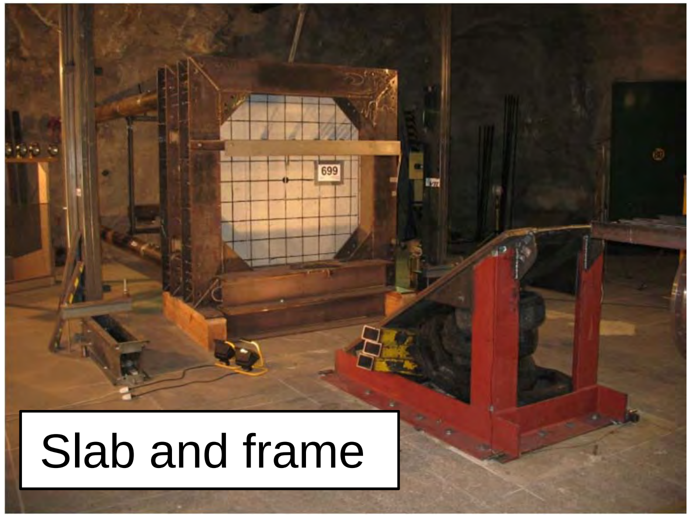
Target slab



Hard missile



Reinforcement



Slab and frame

HIGH SPEED VIDEOS OF HARD MISSILE IMPACT

TEST 699

- Missile mass ca. 47 kg
- Impact velocity ca. 100 m/s

FRONT SIDE

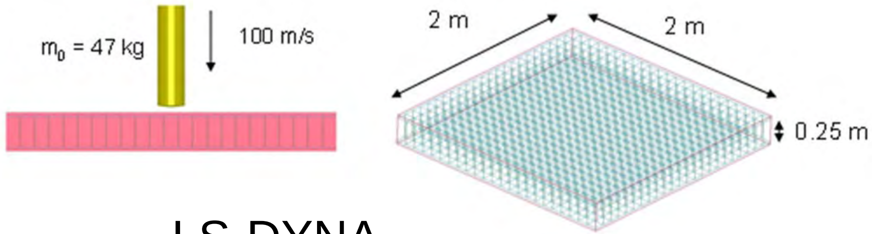
BACK SIDE

SELECTED TEST 699 RESULTS

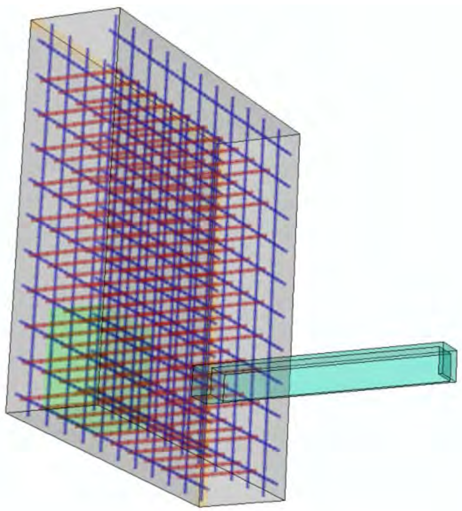
- Penetration (ca. 38 mm)
- Punching cone (ca. 45°)
- Scabbing (circular, Ø ca. 700 mm)

ANALYSIS OF TEST 699

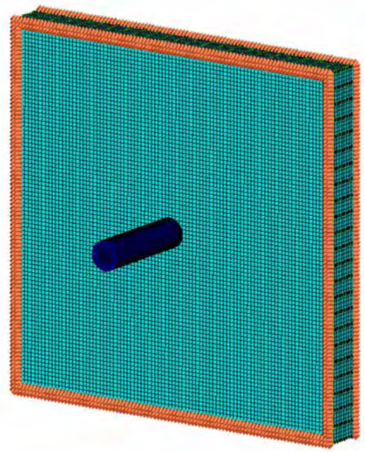
- Complex FE computer codes (user)
  - LS-DYNA (IRSN)
  - ANSYS AUTODYN (GRS)
  - Abaqus/Explicit (VTT)
- Empirical formula used by TUT
  - Penetration depth
  - Perforation limit
  - Scabbing limit



LS-DYNA



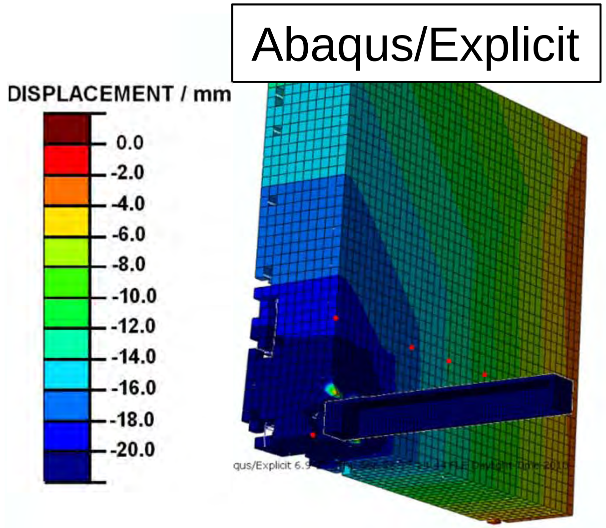
Abaqus/Explicit

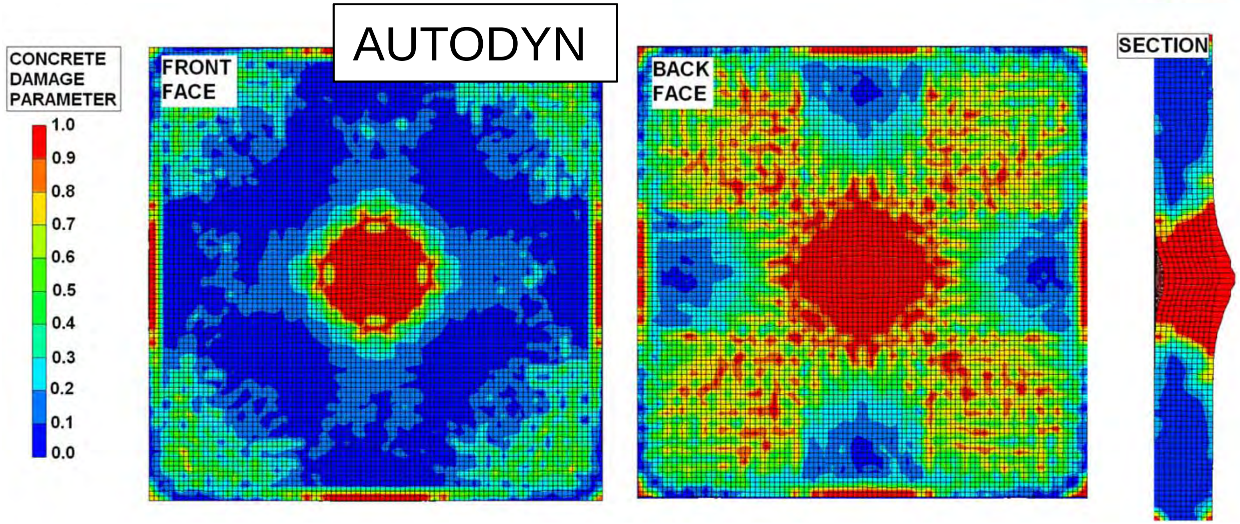


AUTODYN

TEST 699 ANALYSIS RESULTS

Computer codes reproduce
- Penetration
- Punching cone cracking
- Scabbing



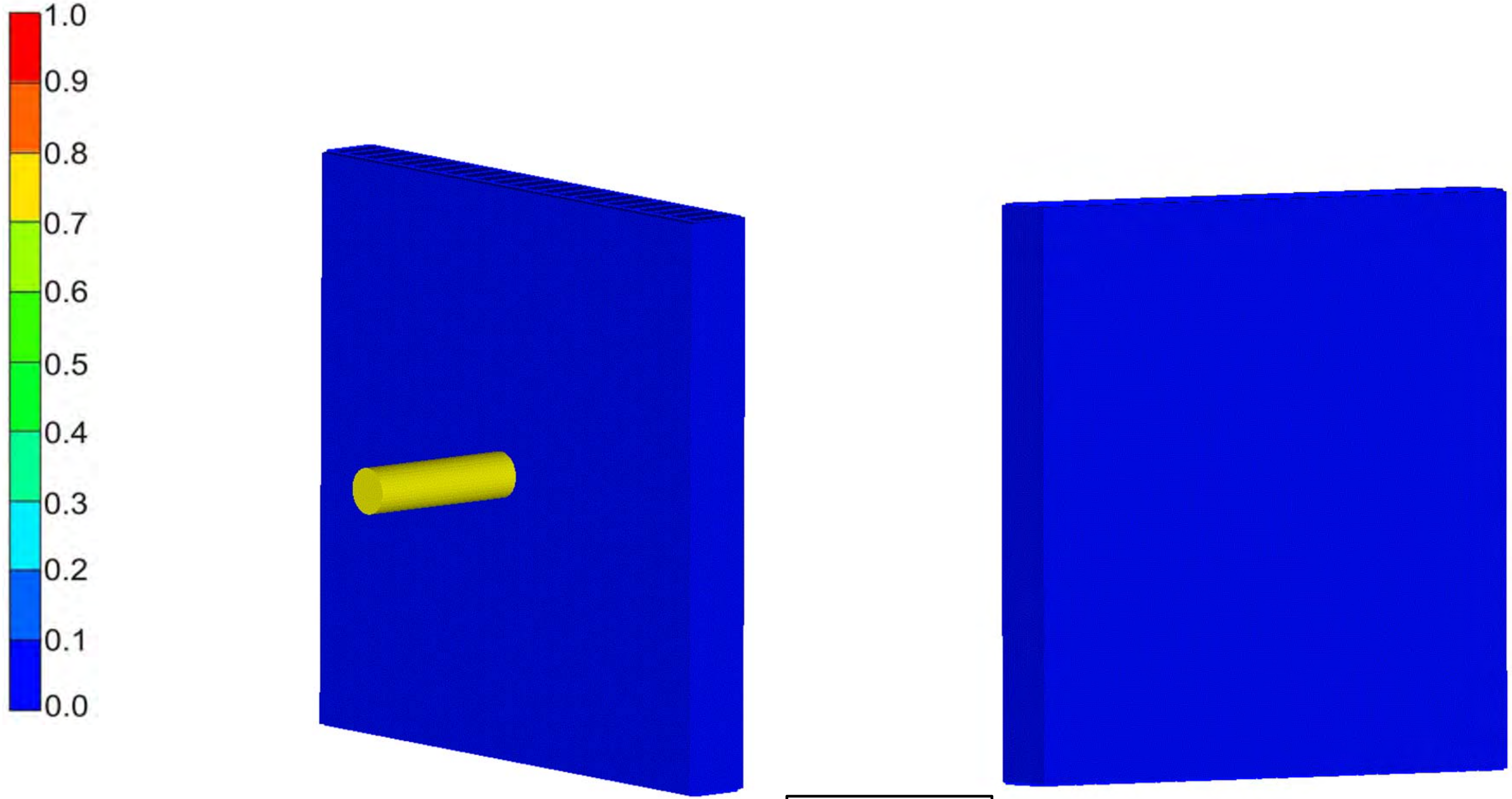


TEST 699 AUTODYN DAMAGE RESULTS

Concrete damage parameter

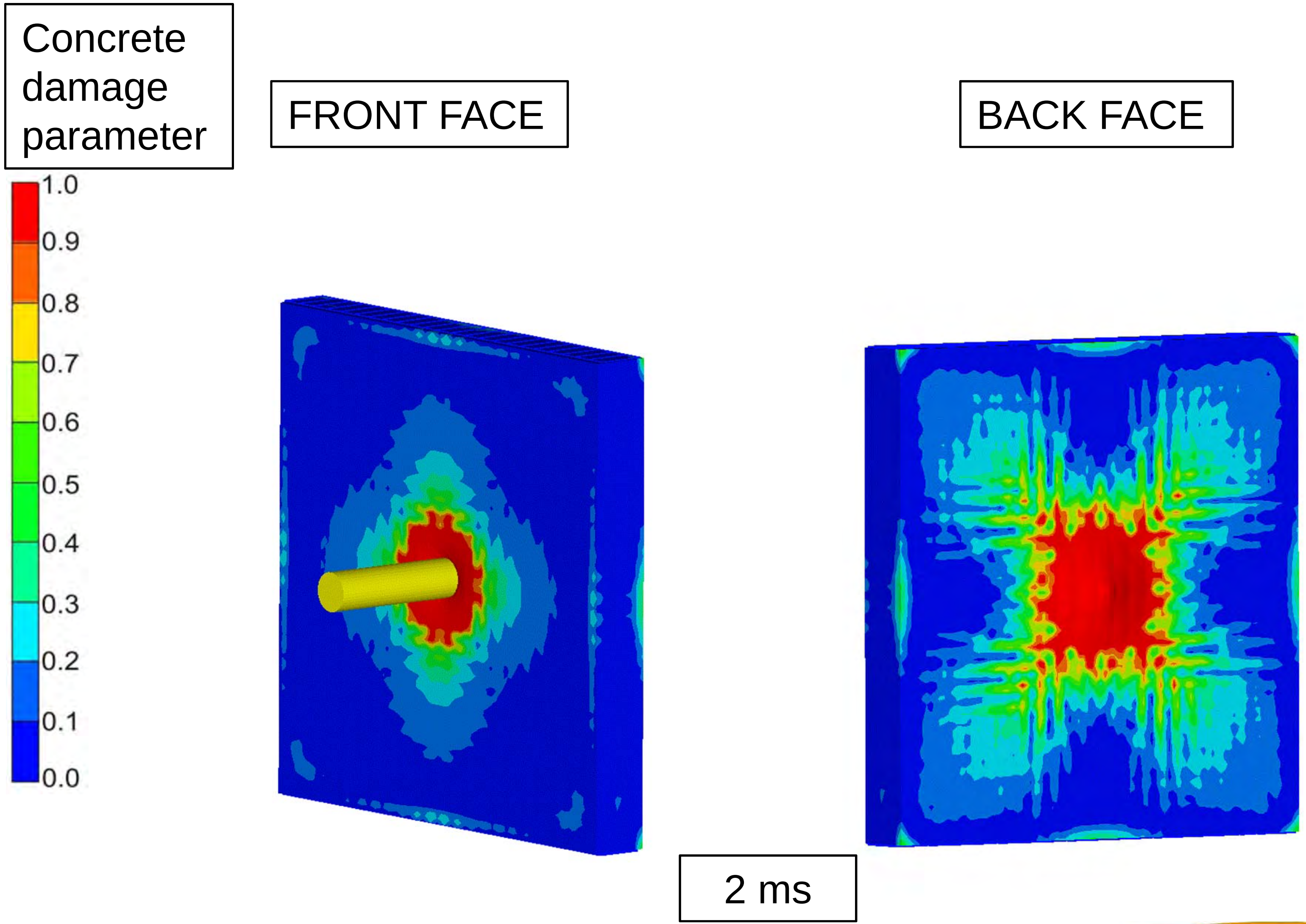
FRONT FACE

BACK FACE



0 ms

TEST 699 AUTODYN DAMAGE RESULTS

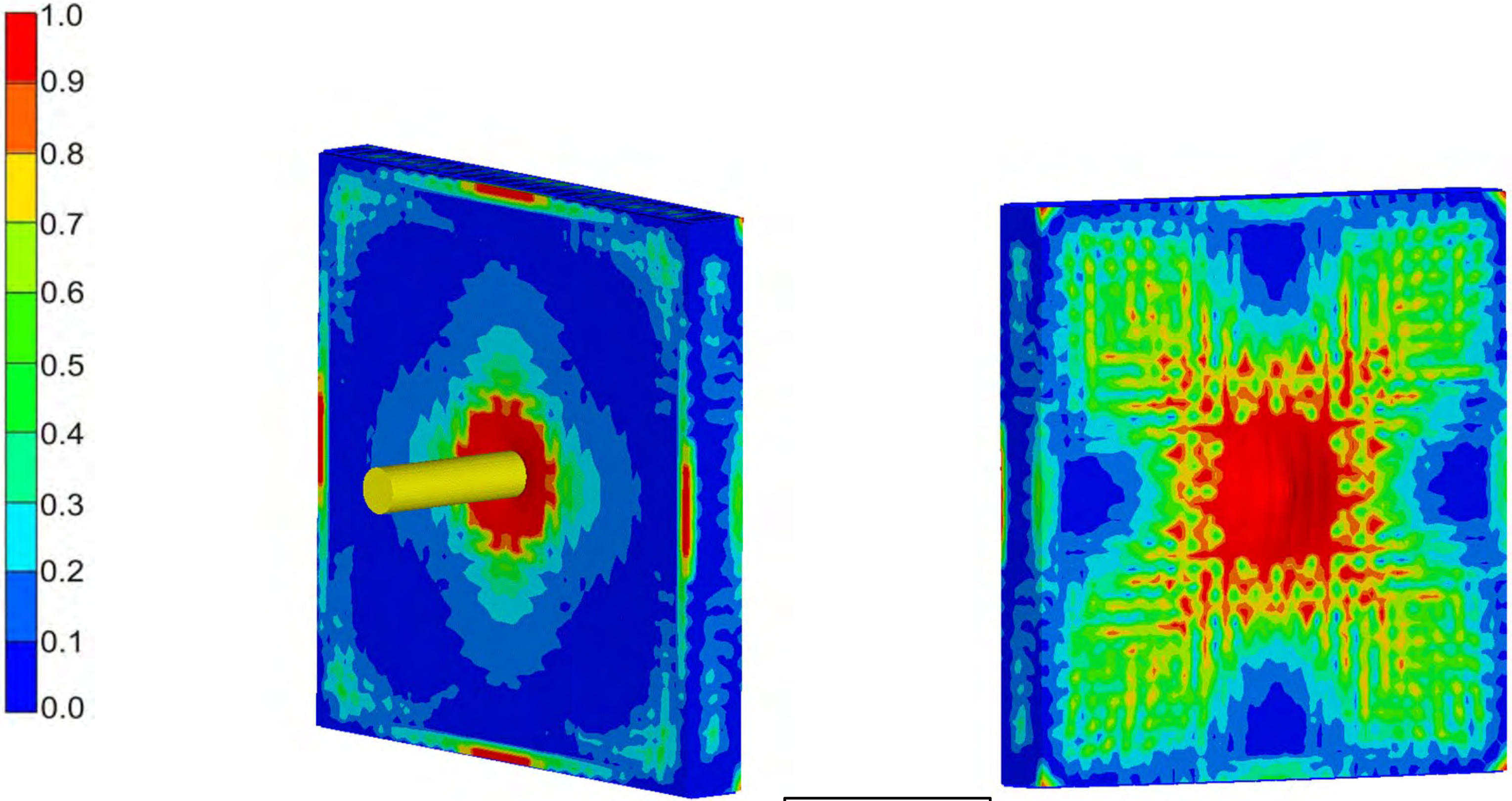


TEST 699 AUTODYN DAMAGE RESULTS

Concrete damage parameter

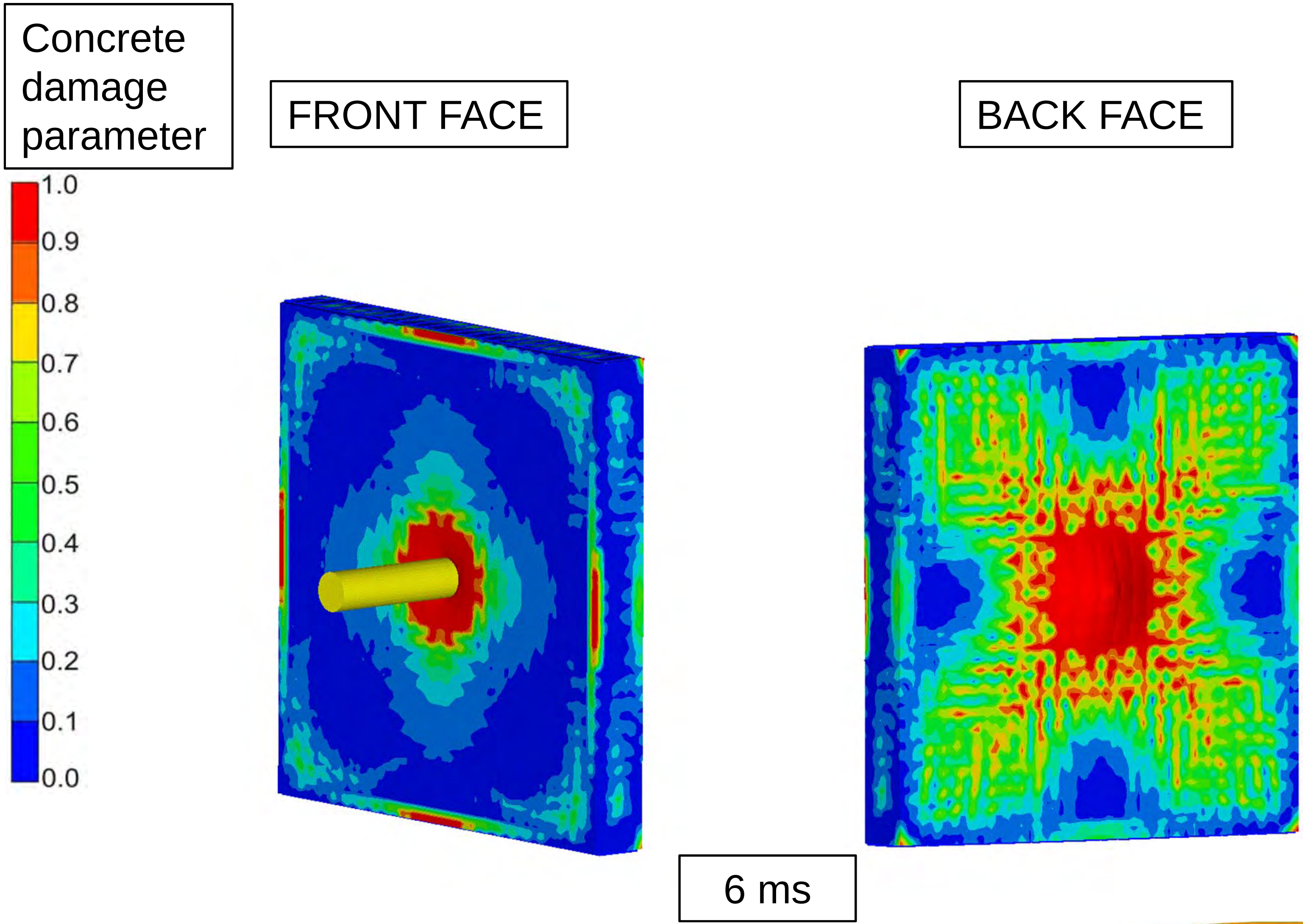
FRONT FACE

BACK FACE

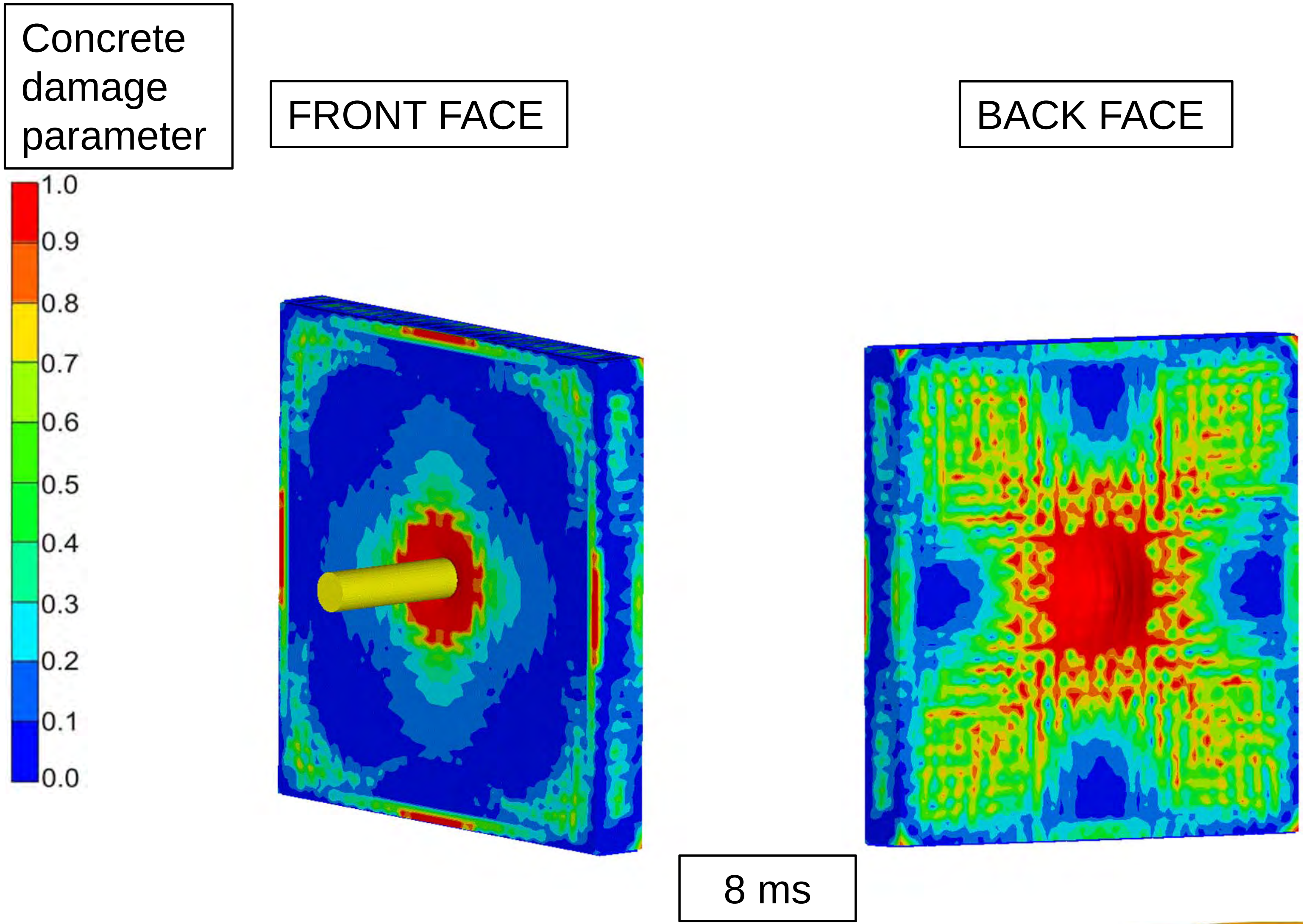


4 ms

TEST 699 AUTODYN DAMAGE RESULTS



TEST 699 AUTODYN DAMAGE RESULTS

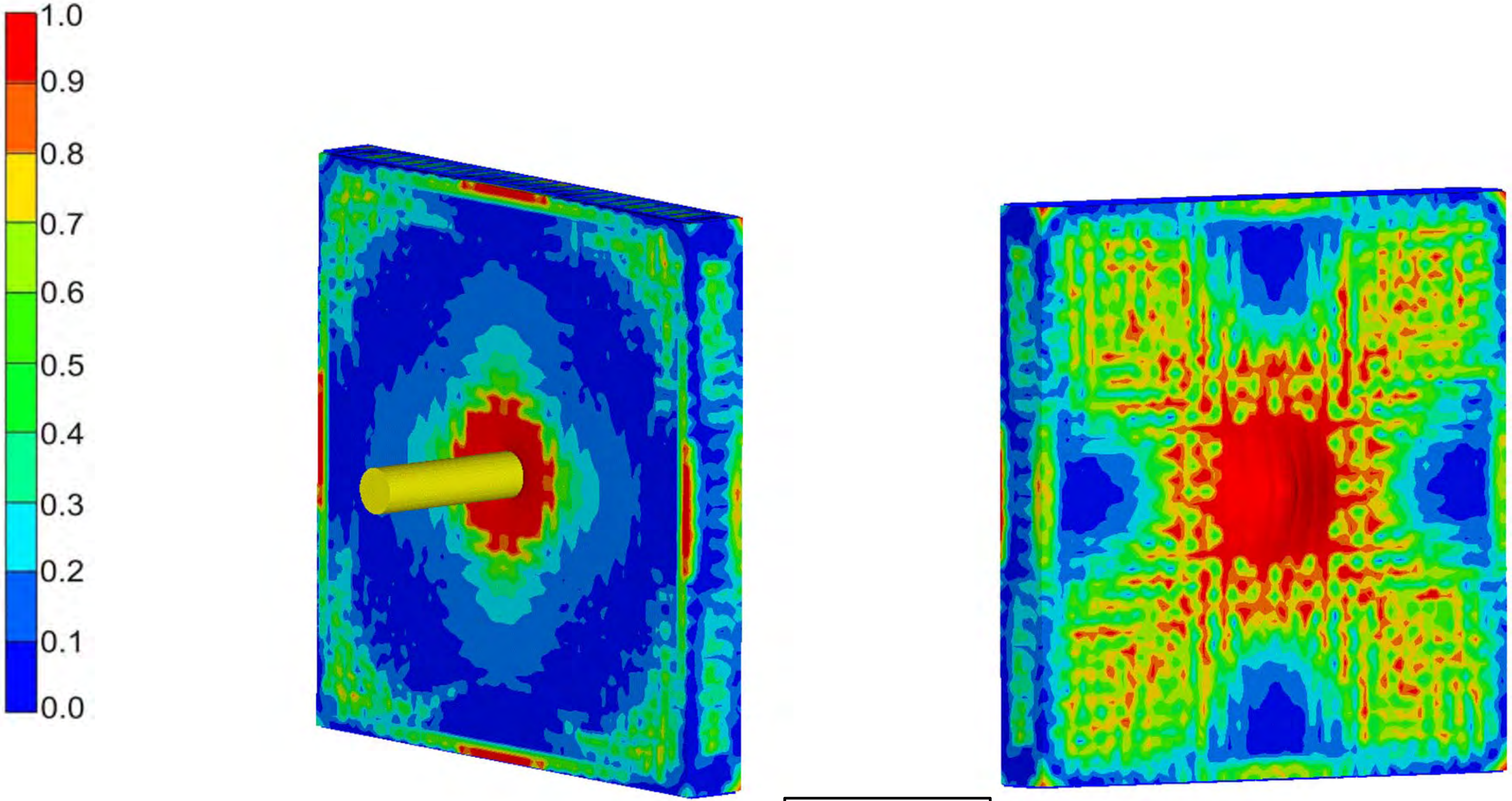


TEST 699 AUTODYN DAMAGE RESULTS

Concrete damage parameter

FRONT FACE

BACK FACE



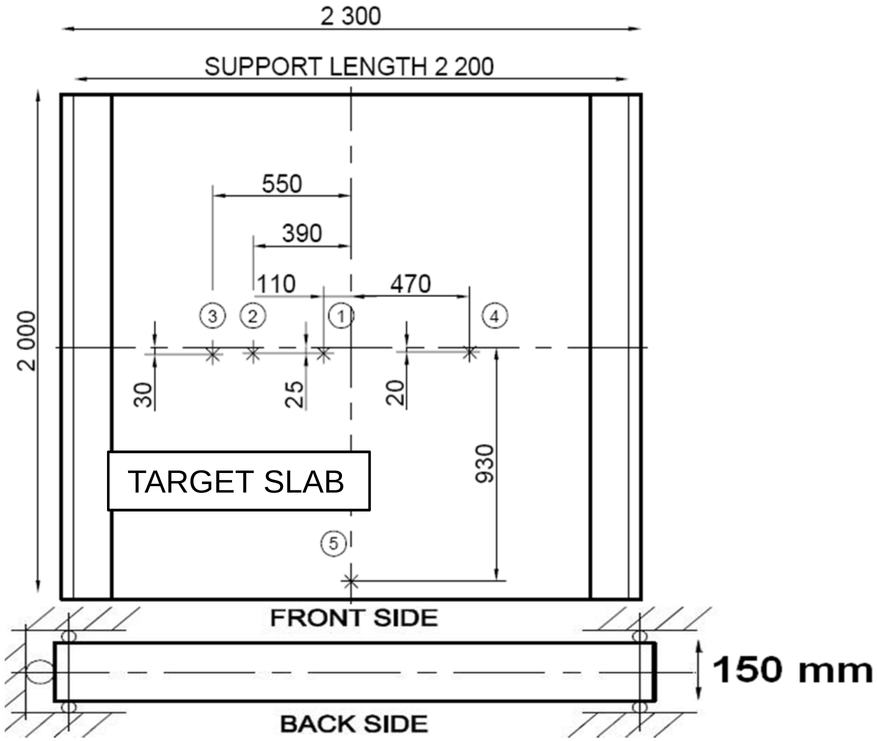
10 ms

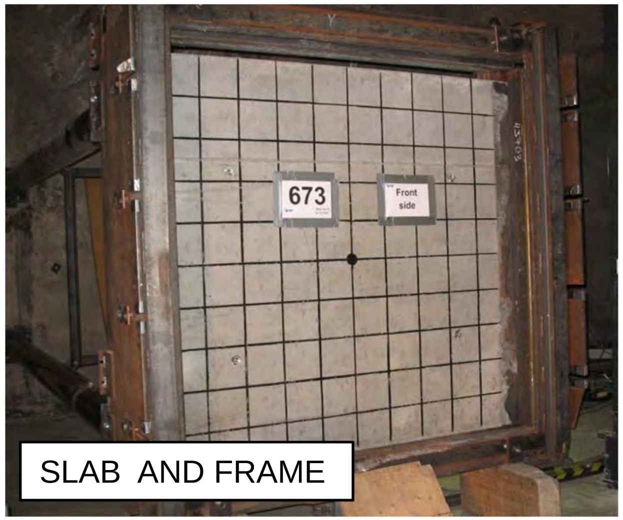
TEST 699 – COMPARISON OF SELECTED RESULTS

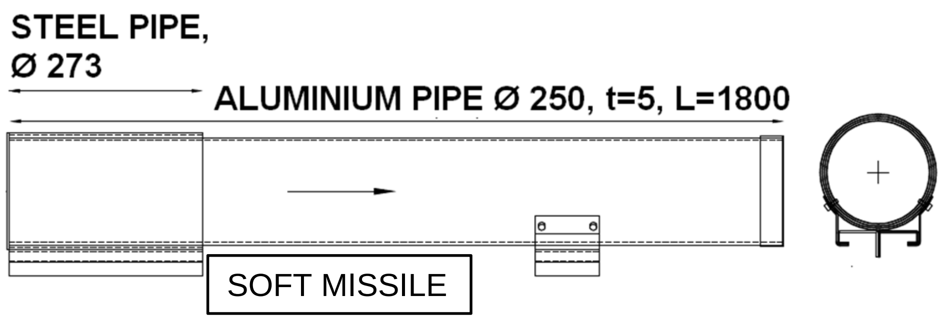
	Tool	Penetration depth	Scabbing diameter	Cone angle
IRSN	LS-DYNA	14 mm	1100 mm	about 45°
GRS	AUTODYN	ca. 40 mm	600-700 mm	about 45°
VTT	Abaqus/Explicit	40-60 mm	600-800 mm	about 45°
TUT	Simplified methods	25-142 mm	-	-
Test	-	38 mm	700 mm	about 45°

- Scattering among the simulations
- Dependencies on input parameters
- Scattering of test results

SOFT MISSILE TEST SETUP (TEST 673)







HIGH SPEED VIDEOS OF SOFT MISSILE IMPACT

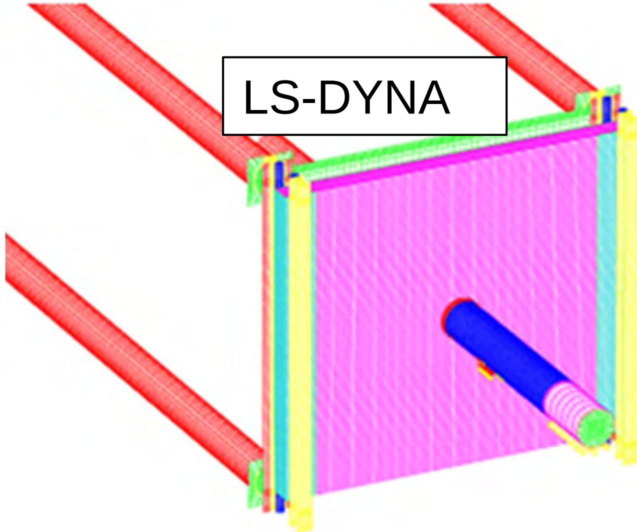
TEST 673
- Missile mass ca. 50 kg
- Impact velocity ca. 127 m/s

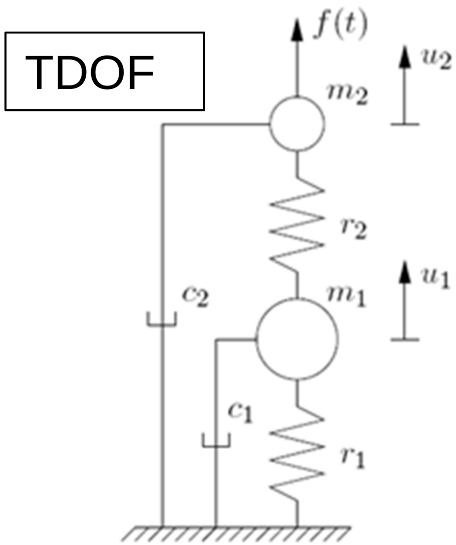
FRONT SIDE

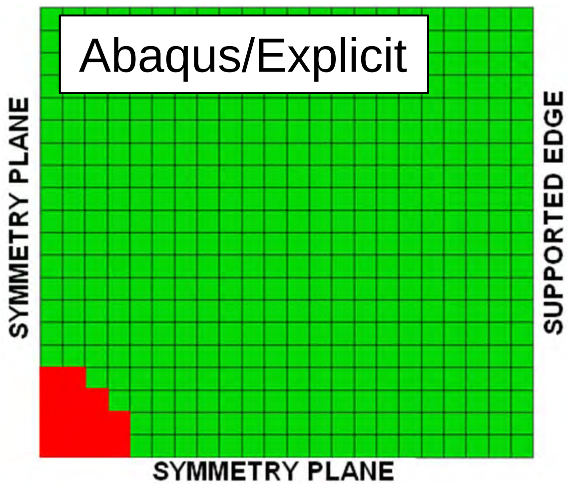
BACK SIDE

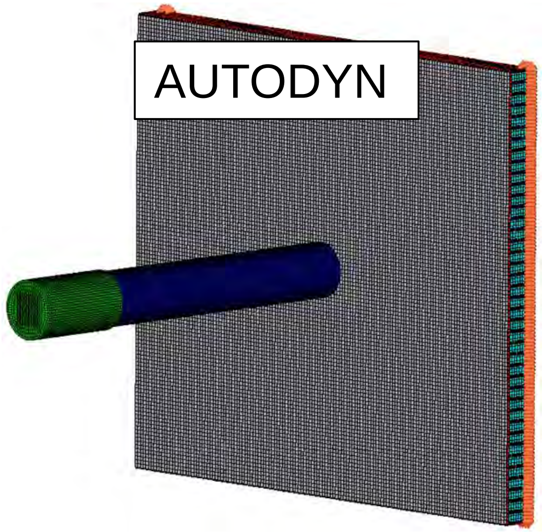
ANALYSIS MODELS

- Complex FE computer codes (user)
  - LS-DYNA (IRSN)
  - ANSYS AUTODYN (GRS)
  - Abaqus/Explicit (VTT)
- Simplified TDOF model (TUT)









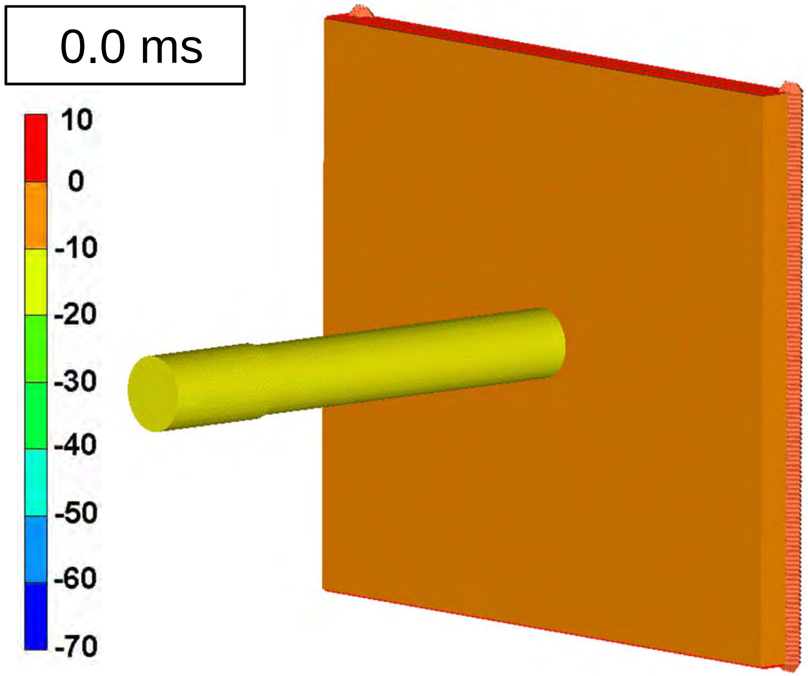
TEST 673 ANALYSIS RESULTS

Test result and fundamental simulation results:

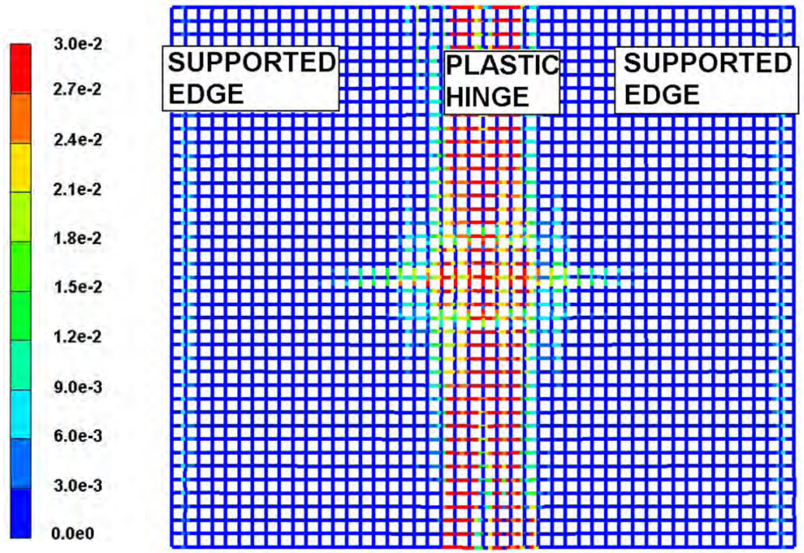
- Flexural vibration of the slab
- Plastic hinge at mid span
- No indication of punching or perforation

SLAB DISPLACEMENTS / mm

0.0 ms



PLASTIC STRAINS BACK FACE REBARS / mm/mm



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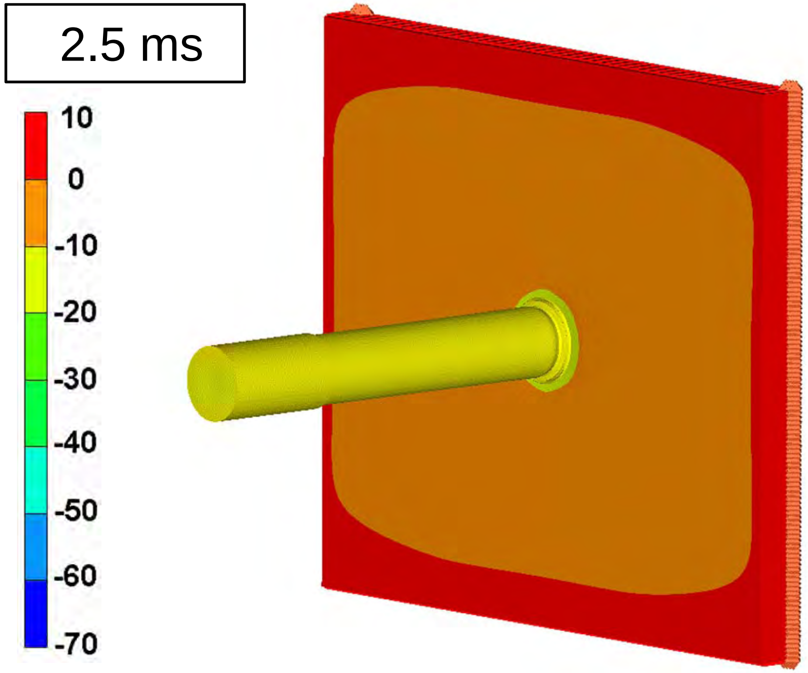

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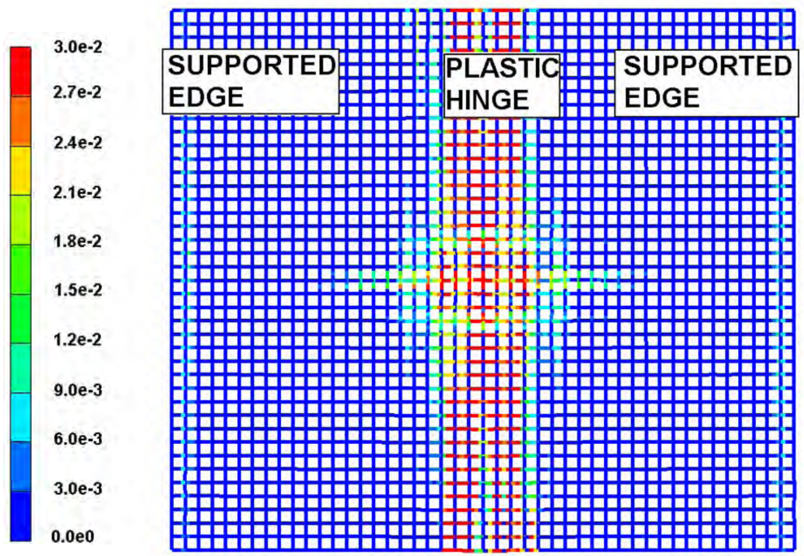

TEST 673 ANALYSIS RESULTS

Test result and fundamental simulation results:

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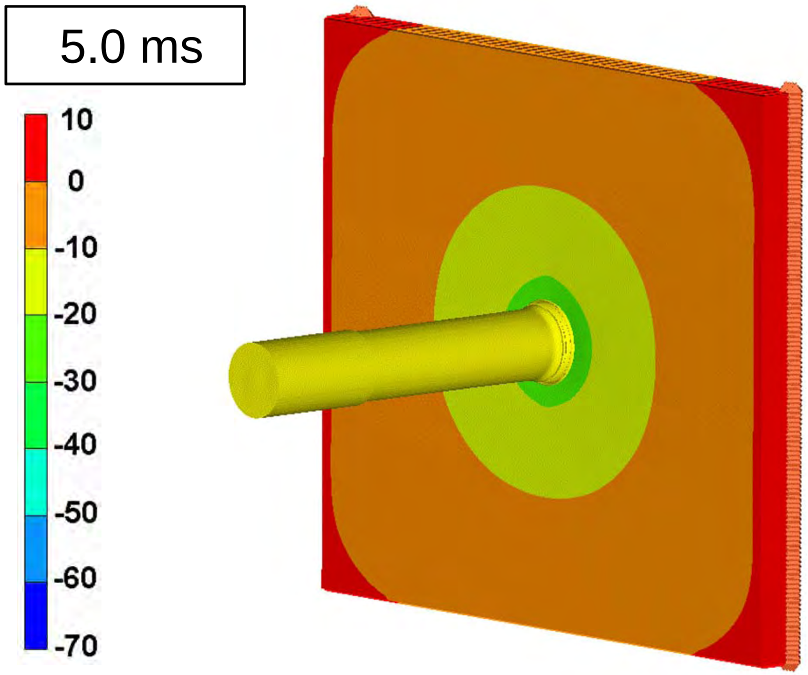
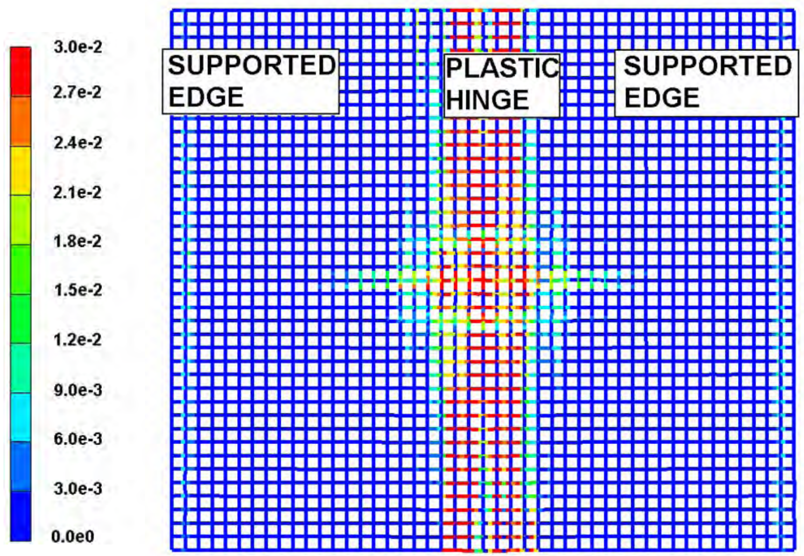
- Plastic hinge at mid span
- No indication of punching or perforation

SLAB DISPLACEMENTS / mm

5.0 ms



PLASTIC STRAINS BACK FACE REBARS / mm/mm



TEST 673 ANALYSIS RESULTS

Test result and fundamental simulation results:

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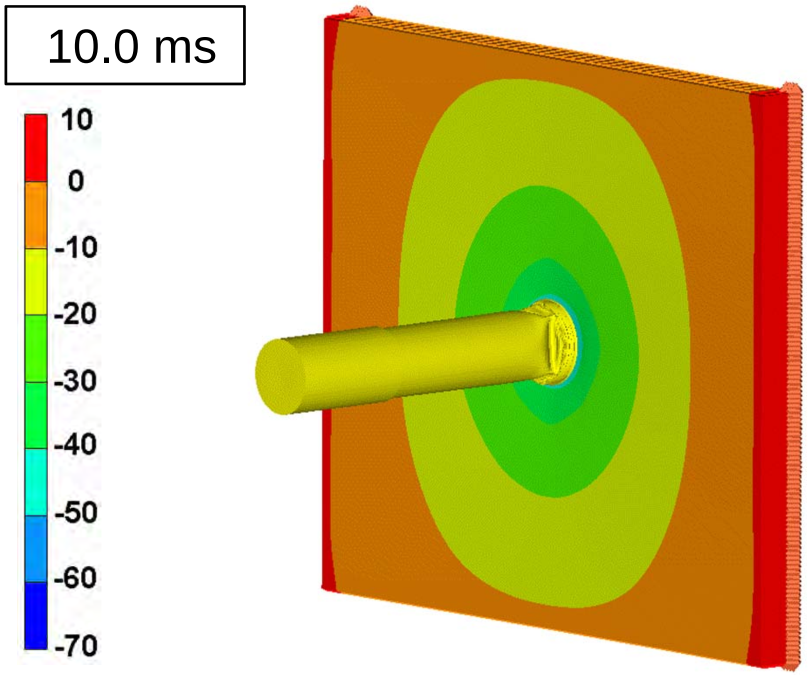
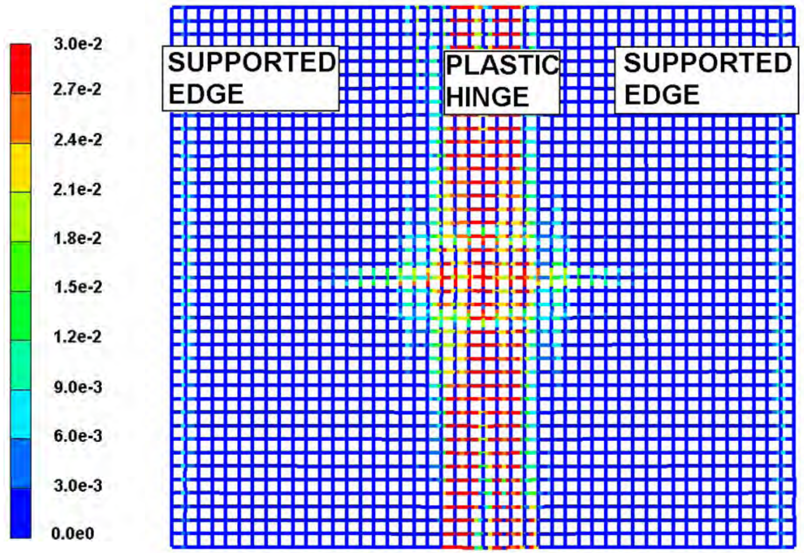
SLAB DISPLACEMENTS / mm

10.0 ms



PLASTIC STRAINS BACK FACE REBARS / mm/mm



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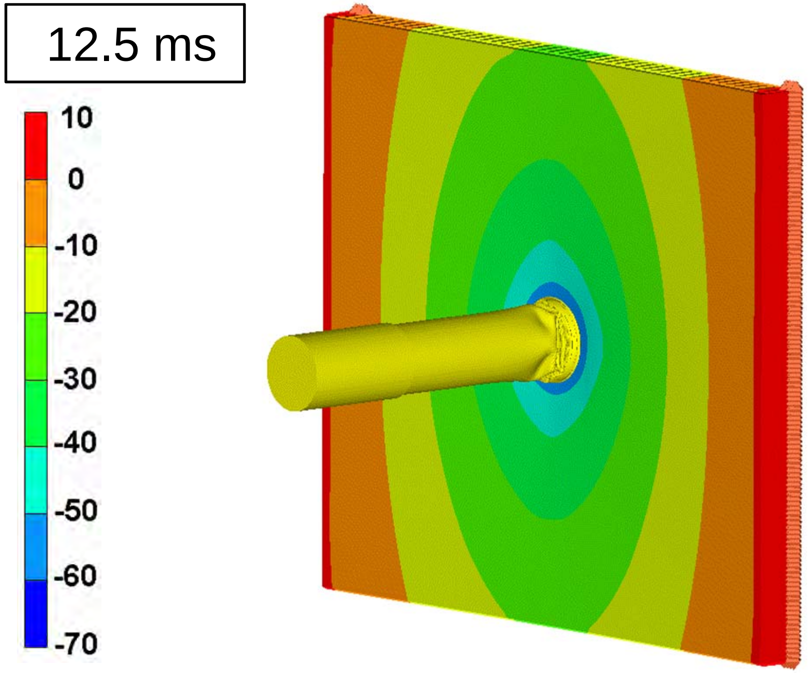
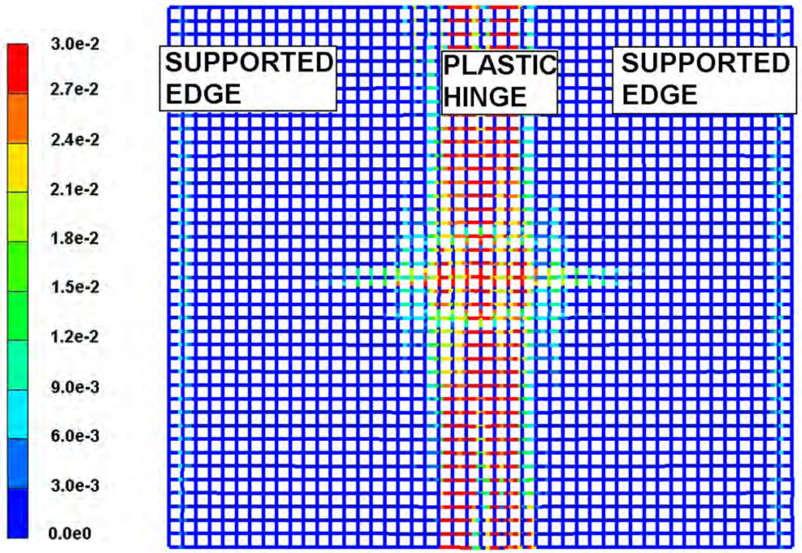


PLASTIC STRAINS BACK FACE REBARS / mm/mm



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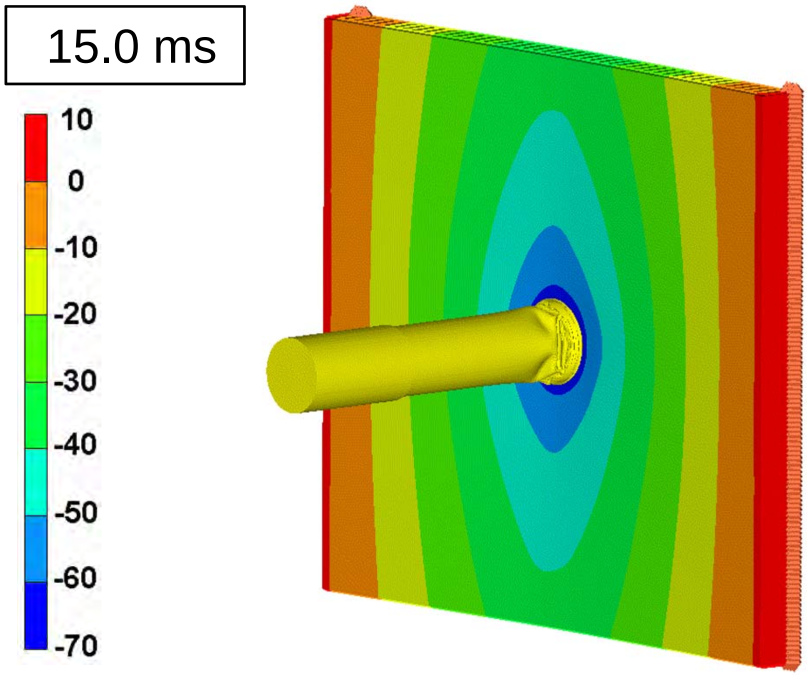

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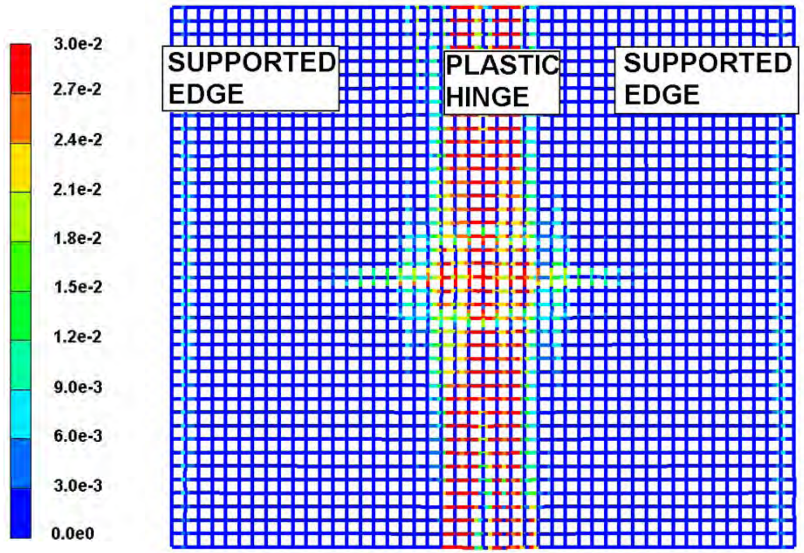

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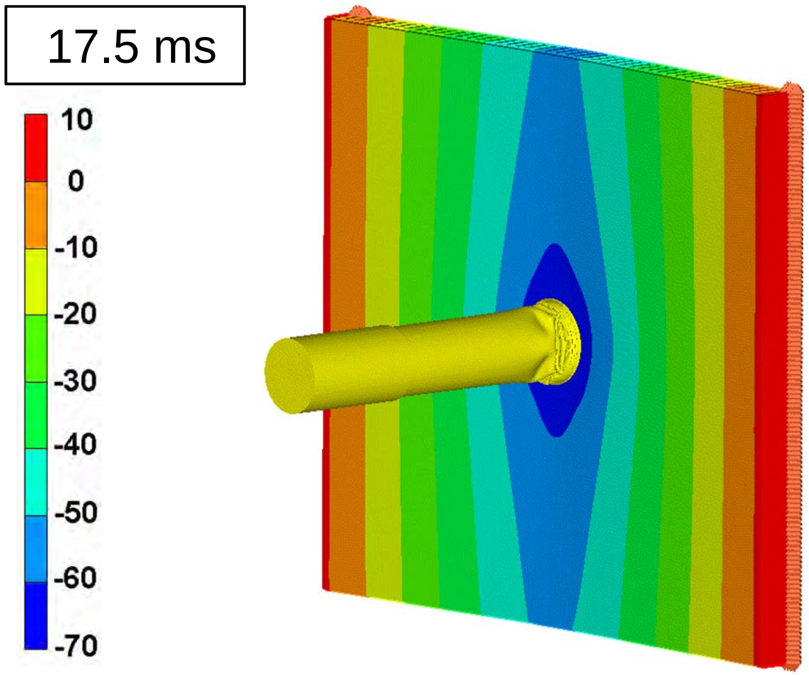
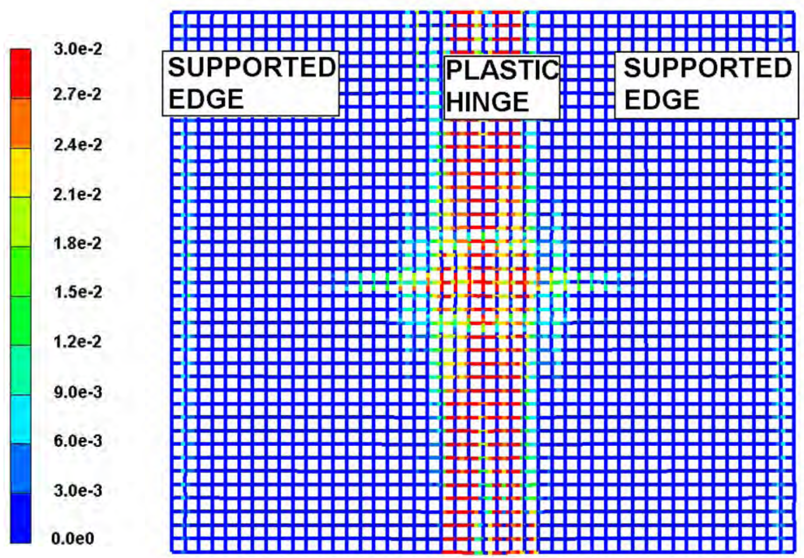


PLASTIC STRAINS BACK FACE REBARS / mm/mm



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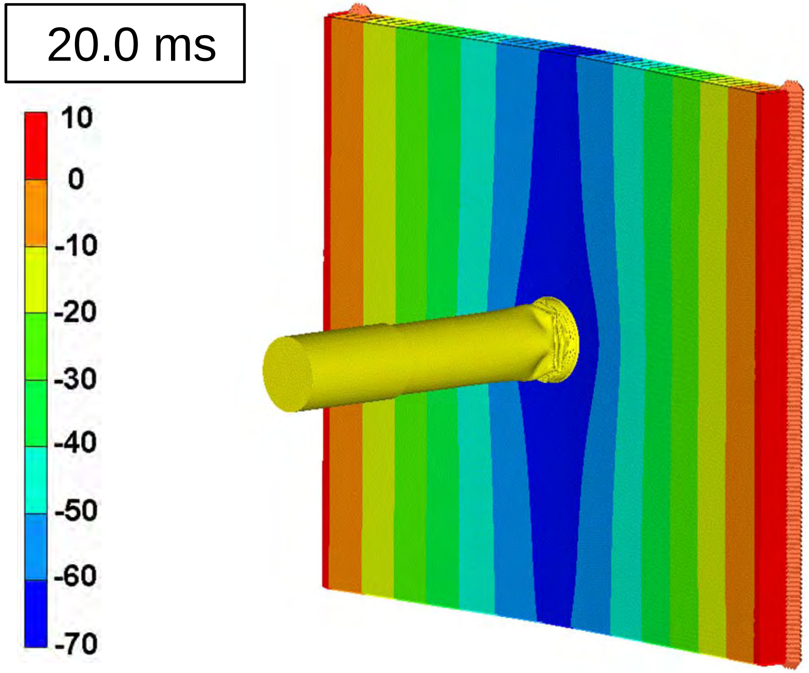

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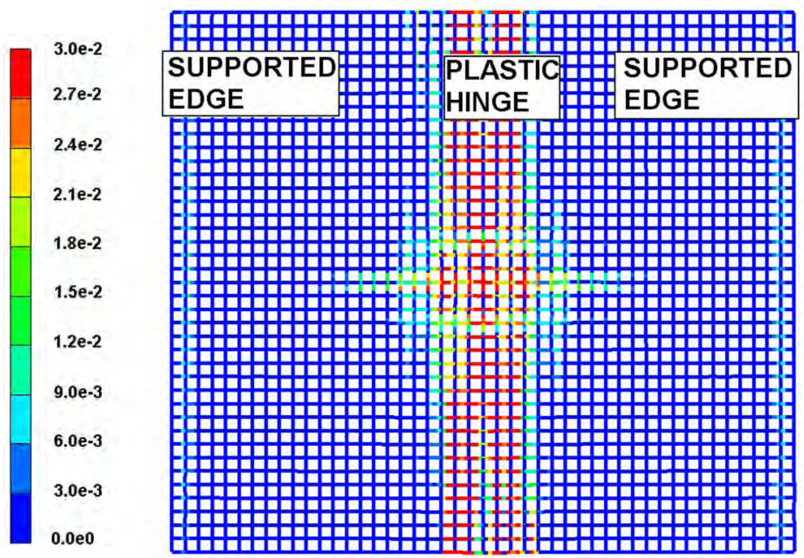

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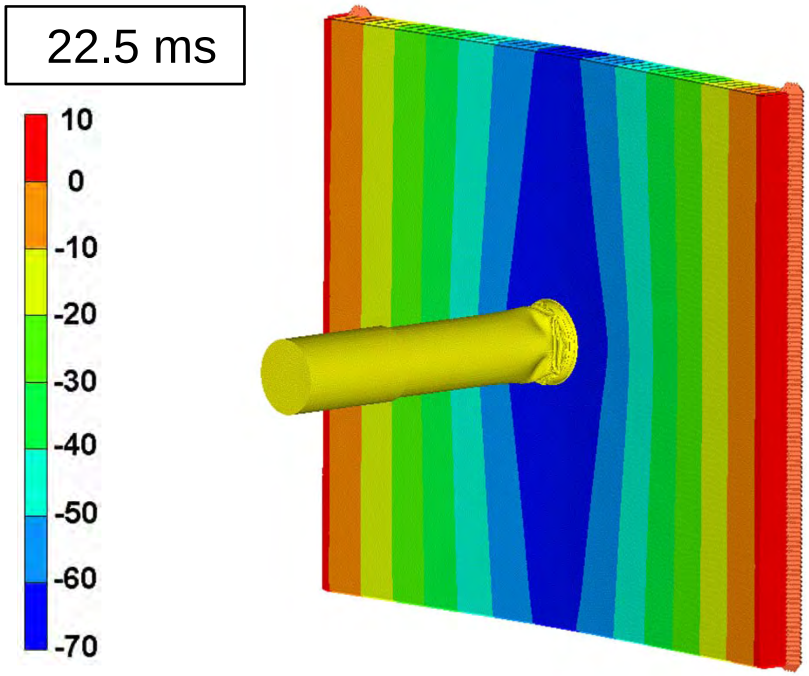

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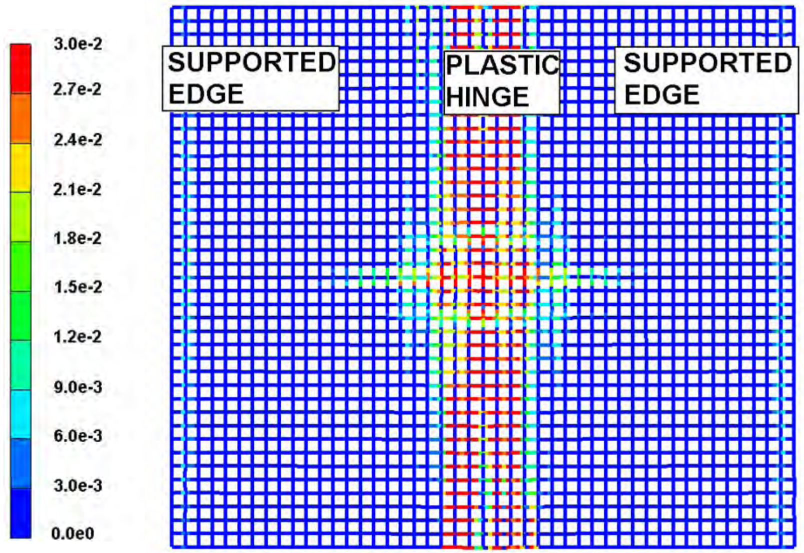

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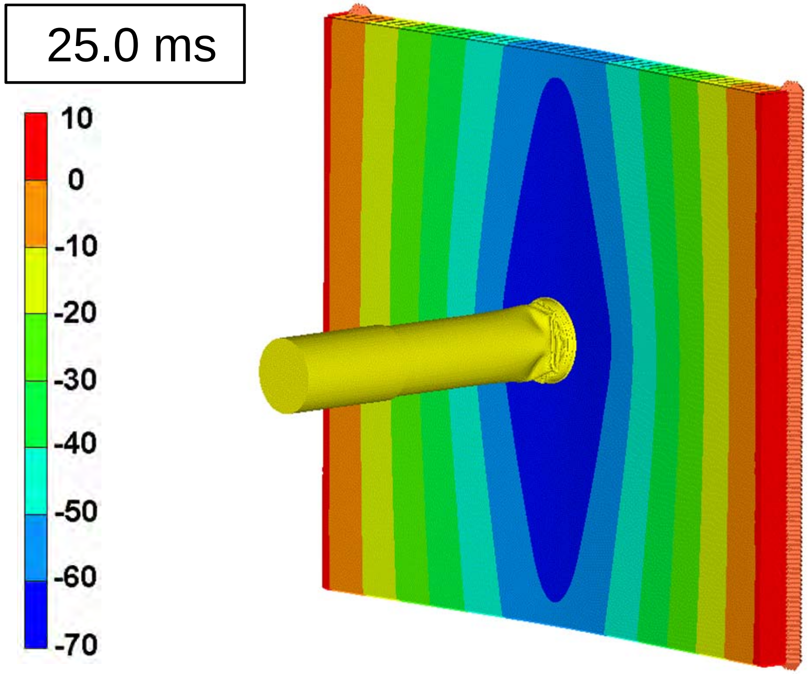
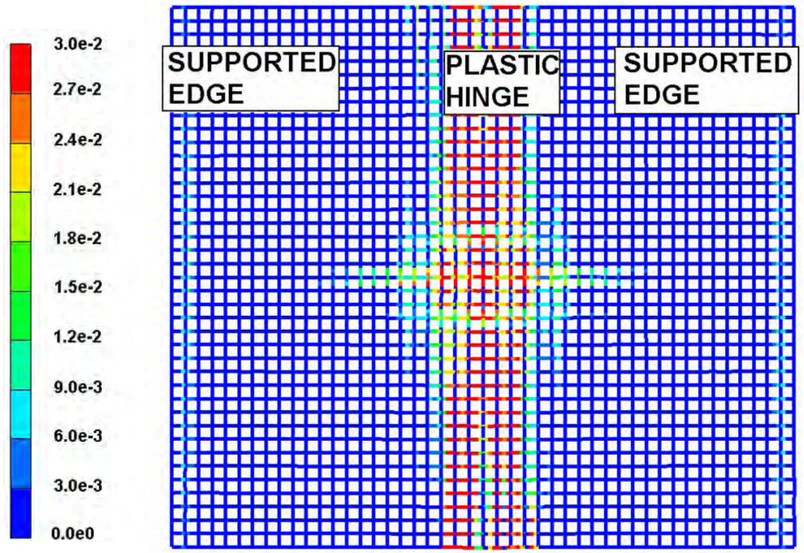


PLASTIC STRAINS BACK FACE REBARS / mm/mm



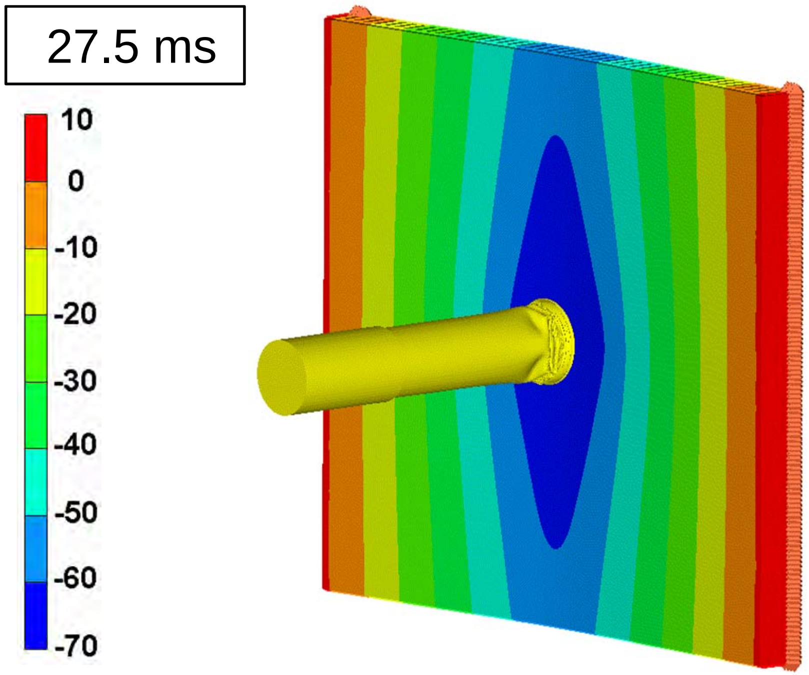
TEST 673 ANALYSIS RESULTS

Test result and fundamental simulation results:

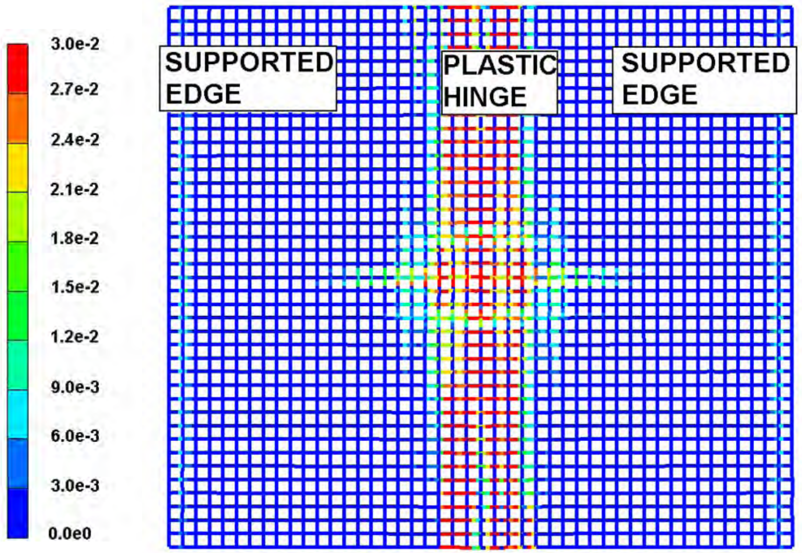
- Flexural vibration of the slab
- Plastic hinge at mid span
- No indication of punching or perforation

SLAB DISPLACEMENTS / mm

27.5 ms

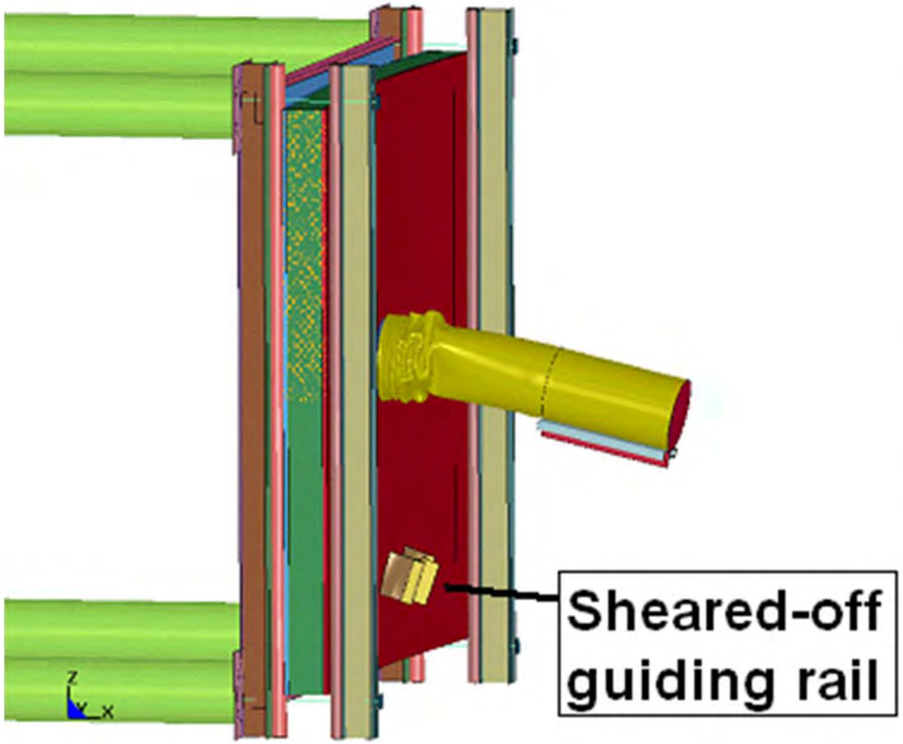


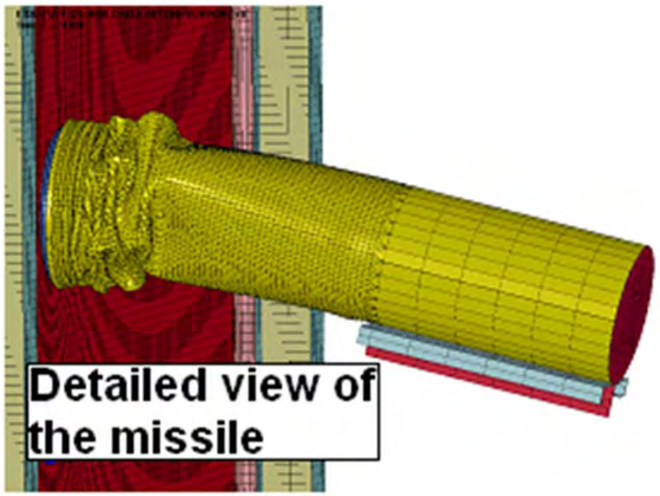
PLASTIC STRAINS BACK FACE REBARS / mm/mm



TEST 673 ANALYSIS RESULTS

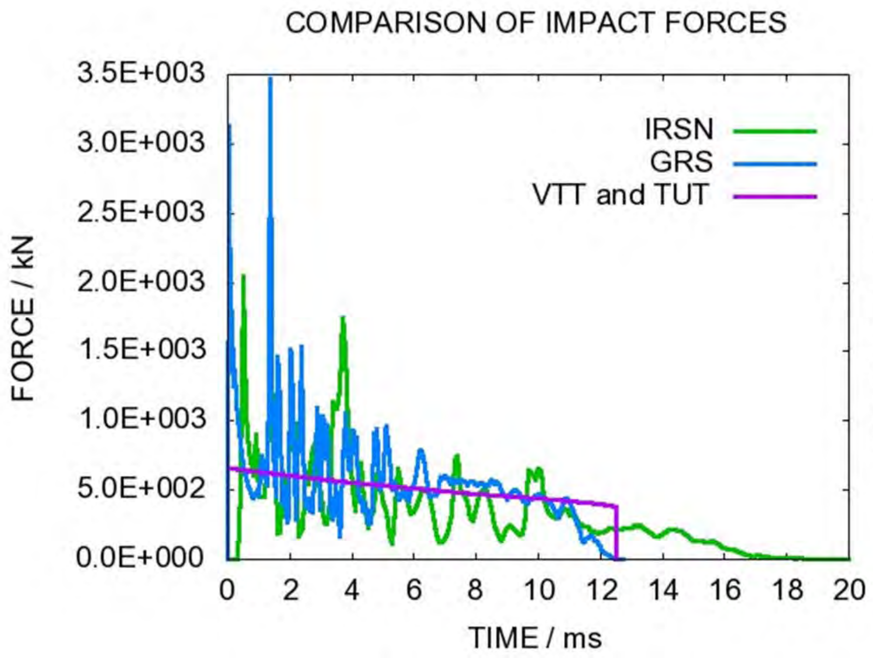
Selected LS-DYNA results

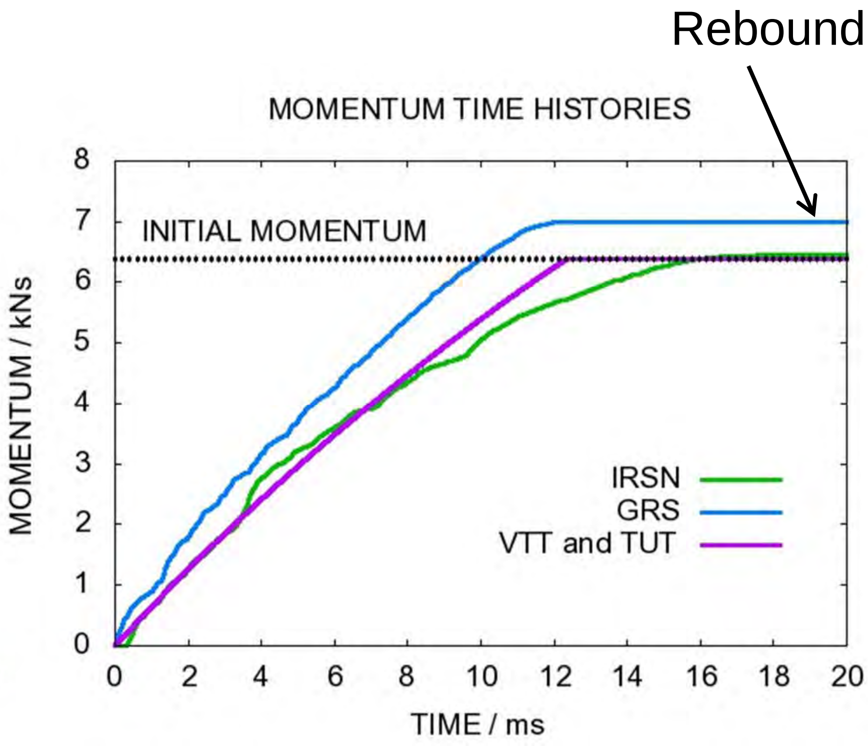




IMPACT FORCES

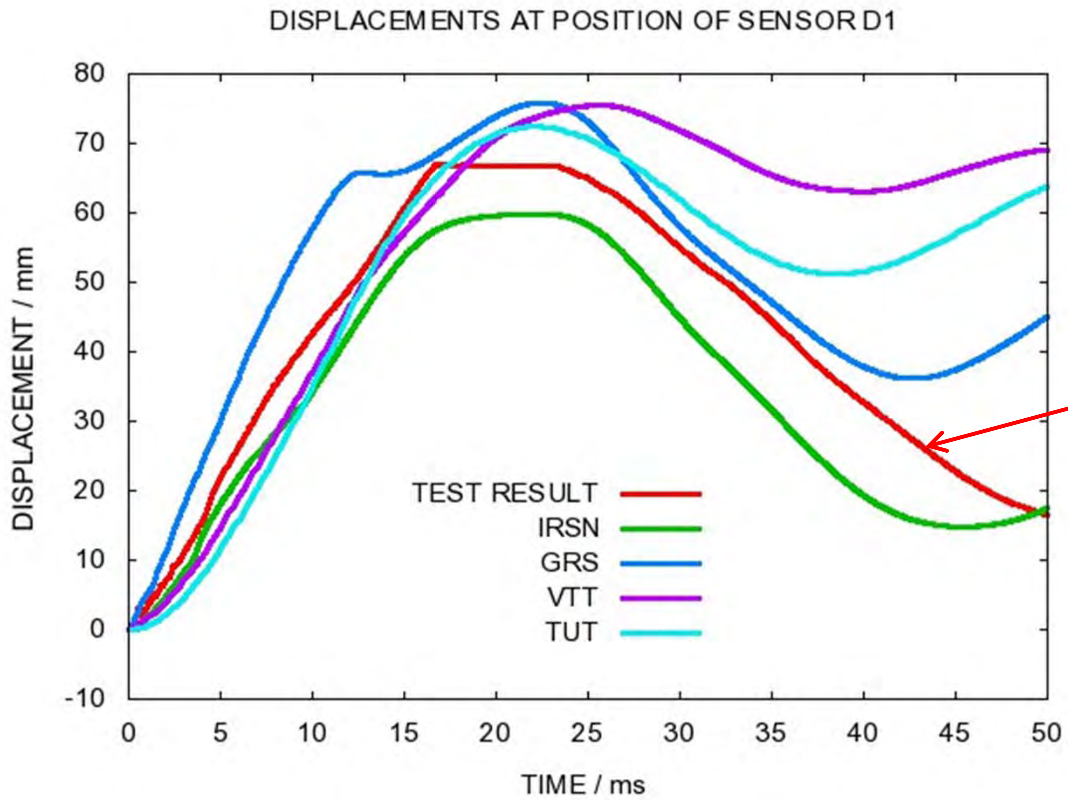
- Calculated forces vs. simplified loading assumption based on Riera model
- Comparison of momentum as integrated contact forces

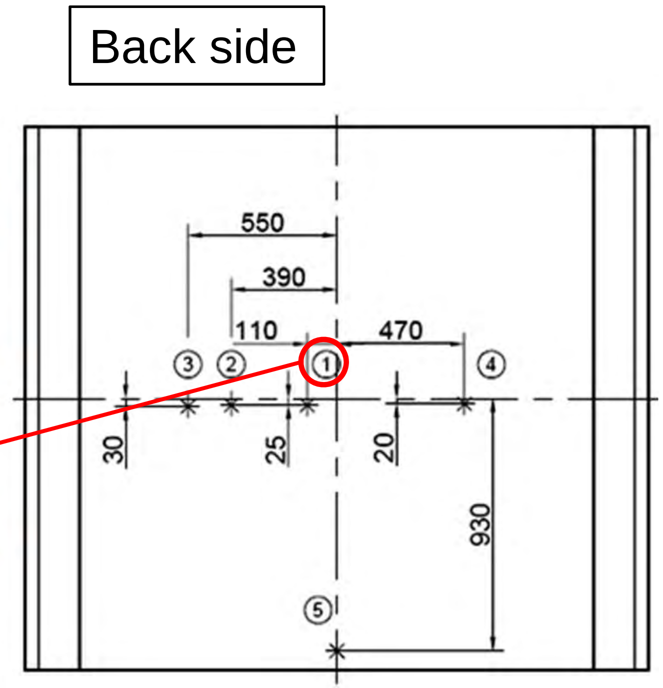




SLAB DISPLACEMENTS

- Maximum displacement well reproduced
- Deviations concerning frequency
- Simplified model useful for sensitivity studies





SUMMARY

- Comparative simulations of VTT impact tests show
  - Employed analysis methods suitable to predict mechanical behaviour of concrete structures
  - Scattering of analysis results is a measure for the accuracy of the methods
  - Certain simulation results are quite sensitive to specific input parameters (e.g. concrete material models)
- Further numerical studies carried out in the framework of
  - IMPACT Phase II
  - Benchmark IRIS_2010 hosted by CSNI/IAGE of OECD

Thank you very much for your attention.

Olli Vilkamo; Kirsi Alm-Lytz
Radiation and Nuclear Safety Authority, STUK

Radiation Safety in New Build

STUK - Radiation and Nuclear Safety Authority

Mission:

Protecting people, society, environment, and future generations from harmful effects of radiation

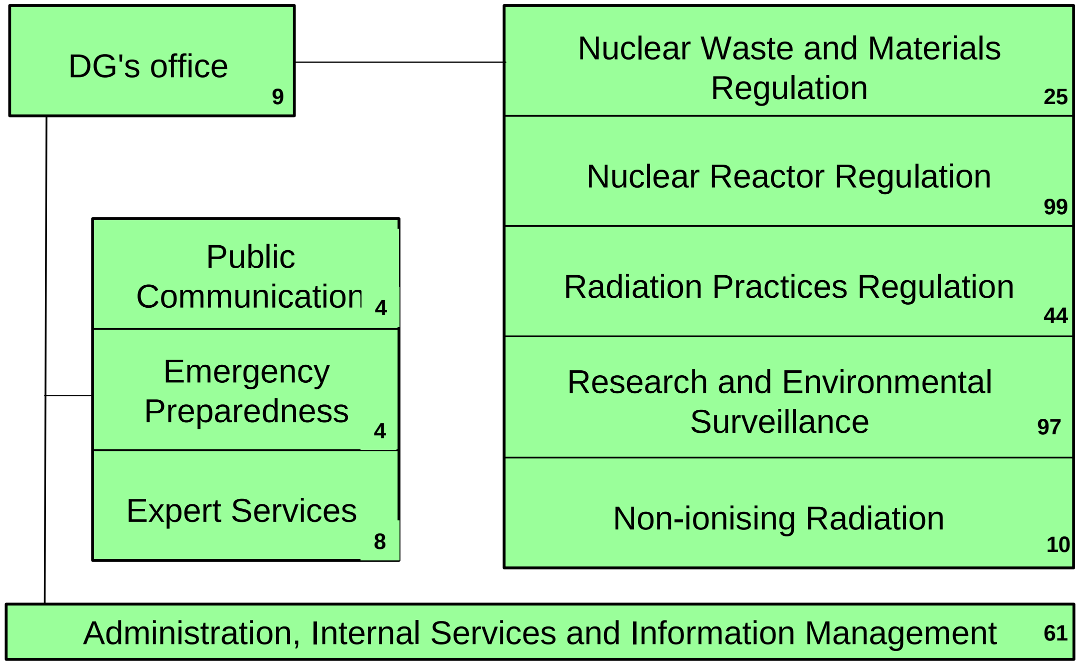
Duties:

Regulation
Research
Services

Offices:

Helsinki
Rovaniemi
NPP site inspectors

Organisation:



Figures indicated number of staff in 2008. About 360 still a total in 10/2010, Nuclear regulation more than 1/3 of activities.

Outline

- Licensing of a NPP in Finland
- Regulatory platform on radiation safety
- Review of radiation safety issues in Olkiluoto 3 reactor project

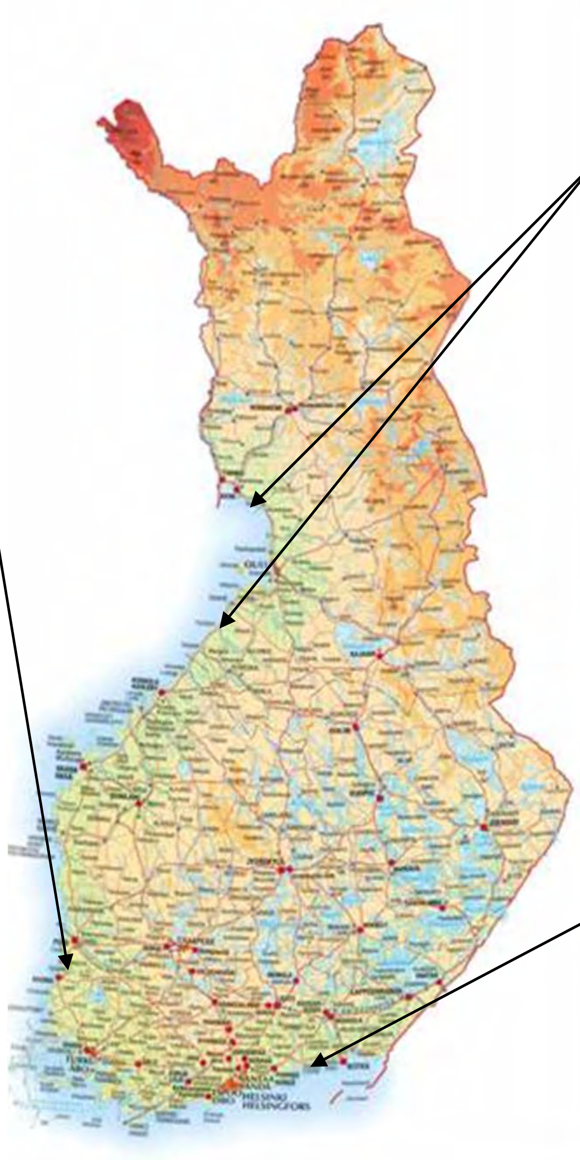
---

- New NPP initiatives and projects, progress until 2010 (preliminary safety assesment, including new site alternatives)

Nuclear power plants in Finland

Olkiluoto NPP (TVO)

- 2 operating units - ABB BWRs
- OL3 (EPR) under construction
- Application for OL4

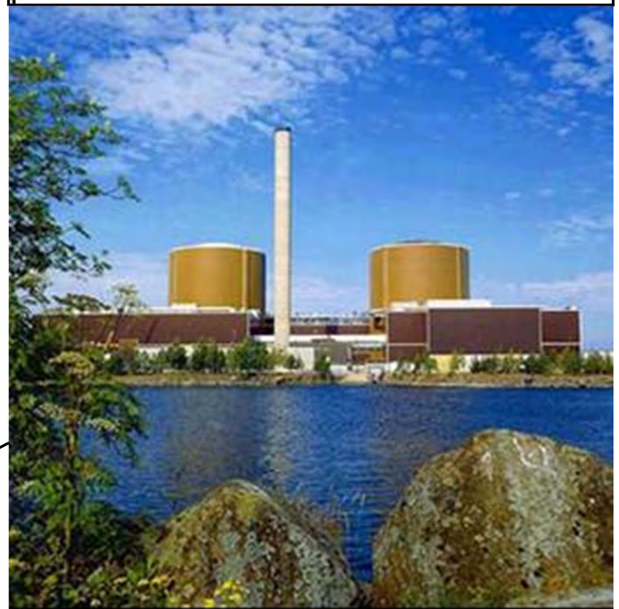


Fennovoima Ltd

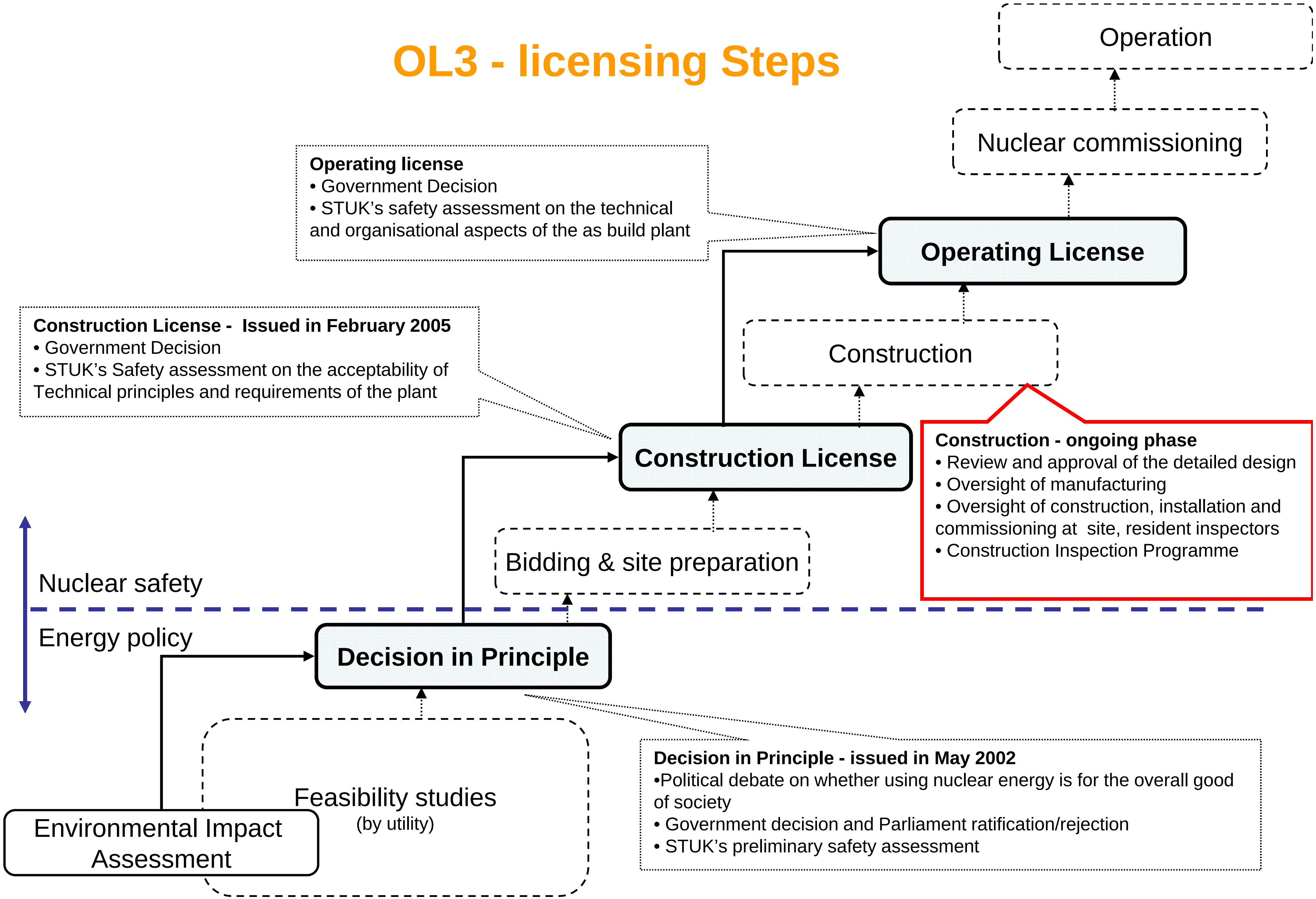
- New utility, no operating reactors, Application for FV1 (2 alternative sites)

Loviisa NPP (Fortum)

- 2 operating units - VVERs
- Application for LO3 (rejected)



OL3 - licensing Steps

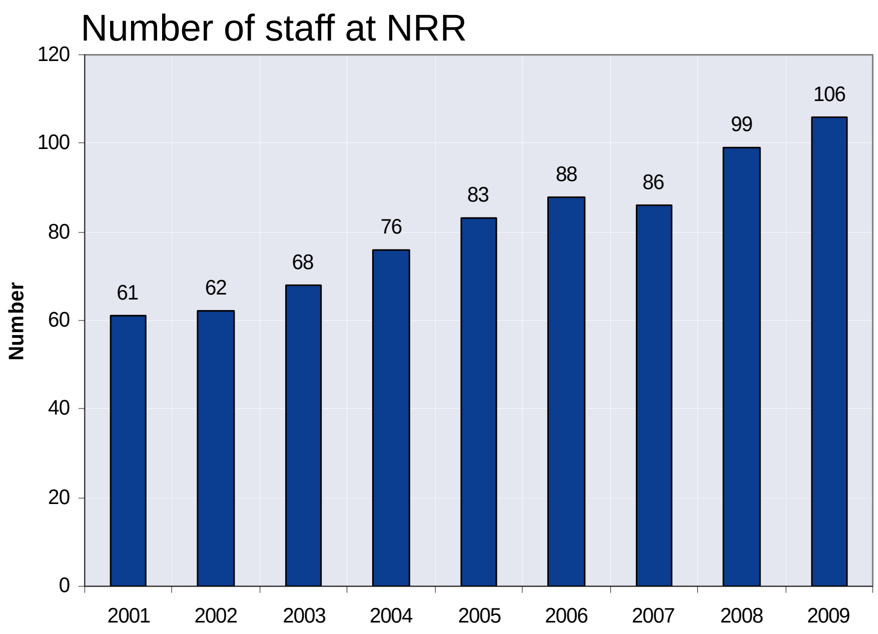


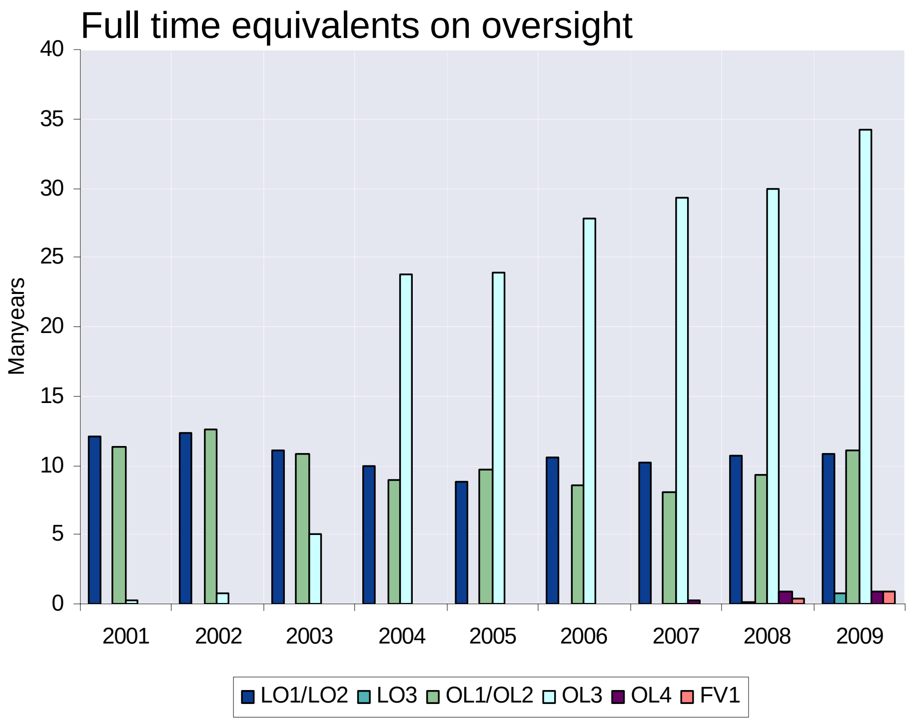
OL3 Project - General

- OL3 is a turn key project; owner and operator (Licensee) is TVO and constructer/vendor is consortium of Areva and Siemens
- Project is progressing but is more than three years behind the original schedule; Lessons learnt, e.g.:
  - Design should be as ready as possible prior construction (no "design as you go")
  - Qualification of a new construction or manufacturing method may take time
  - Key to success is adequate human resources, expertise and experience within all stakeholders (vendor, subcontractor, licensee, regulator)
  - Management of requirements, design configuration and modifications should be systematic

Photos: TVO

STUK's Nuclear Reactor Regulation (NRR), used resources for OL3





Number of staff is currently ~ 110

Radiation safety related YVL Guides

- STUK issues detailed regulations concerning the safety of NPPs in YVL Guides
- Several YVL Guides concern radiation safety issues (siting, **radiation safety aspects in design of NPPs**, radiation protection of workers, radiation monitoring systems, monitoring of discharges, limitation of public exposures, meteorological measurements, radiation monitoring in the environment, emergency preparedness)
- **YVL 7.18, Radiation Safety Aspects in the Design of NPPs (2004):**
  - general design principles: e.g., materials, shielding, design target for collective dose, accident situations
  - radiation safety issues in layout design
  - radiation safety issues in system design
- All YVL Guides are currently under revision, new guides should be published by end of 2011

Radiation safety issues reviewed in the OL3 construction license phase

- sources, lay-out, shielding
- primary system material specification
- primary coolant chemistry
- fuel integrity
- maintenance planning
- collective dose estimation
- on-site habitability during accidents
- releases during normal operation and the use of BAT principle
- accident analyses and radiological consequences
- conceptual design of radiation monitoring system
- waste management issues

Primary system material selection

- Co-60 is typically causing most of workers' doses
- Cobalt hard-facing (stellite) minimisation vs. increase of technical risk and loss of operating experience
- Minimising residual cobalt content in primary system materials, e.g. in stainless steels and SG tubing material
- Stellite and residual cobalt inside the RPV have the largest impact on workers' radiation doses

Target for collective dose (YVL 7.18)

- Design criterion for annual personnel collective dose:

  < 0.5 manSv / 1 GWe averaged **over the plant life**

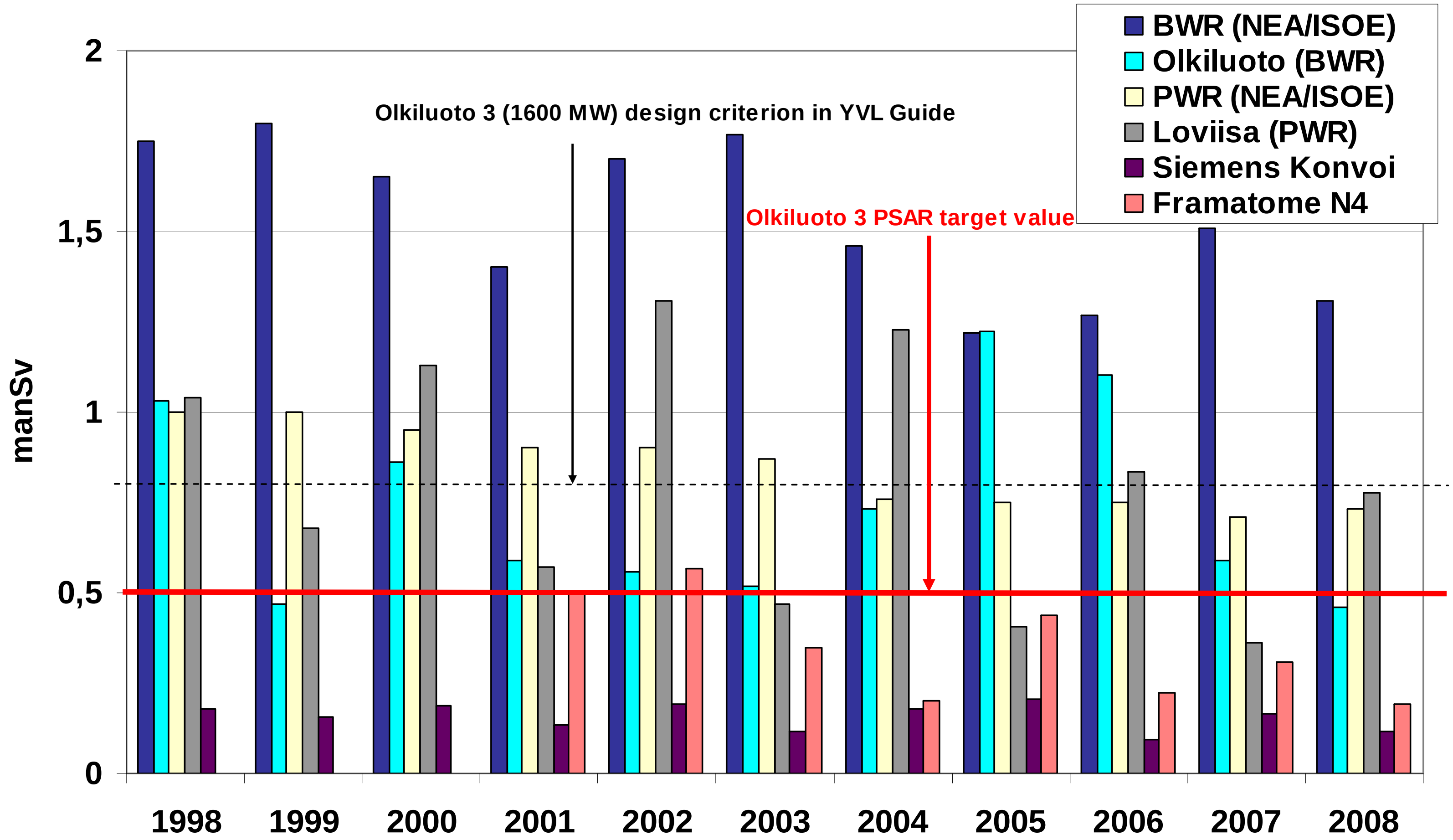
  < 0.8 manSv / year for OL3 (1600 MWe)

- **EUR document** requirement for EPR:
  target for annual collective effective dose averaged over the plant life is **0.5 manSv / year**

- **ALARA principle** should be also applied in the design

- **OL3 (12 months cycle):** 0.43 manSv/year (PSAR phase) / 0.38 manSv/year (more detailed analysis during construction)

Annual collective doses



Limitation of radioactive releases from a nuclear power plant (Government Decree 733/2008 and Regulatory Guide YVL 7.1)

- Acceptance criterion for radioactive releases / max doses to general public for normal operation:
  - **radiation dose limit 0,1 mSv / year for the entire site**
- Radioactive discharges shall be reduced using best available techniques (BAT principle) + ALARA principle
- Anticipated events: **radiation dose limit 0,1 mSv**
- Design basis accidents: **radiation dose limits 1 mSv (class 1) and 5 mSv (class 2)**
- Severe accidents: **no acute health effects, tolerable long term contamination (Cs-137 release < 100 TBq)**

Olkiluoto 3 - discharge abatement

- Operating and R & D experiences from reference plants (N4 & Konvoi) => realistic improvements (BAT)
  - Fuel design
  - Materials in the primary systems
  - Coolant chemistry & purification
  - Gas treatment
  - Liquid waste treatment
  - Ventilation and filtration
- Reference plant values of actual discharges <=> ALARA

On-site habitability during accidents

- Requlatory guide YVL 7.18 requires analyses of the possible radiation sources and estimates of emergency worker doses
- Design stage habitability criterion:
  doses may not exceed the normal dose limits of a radiation worker (50 mSv)

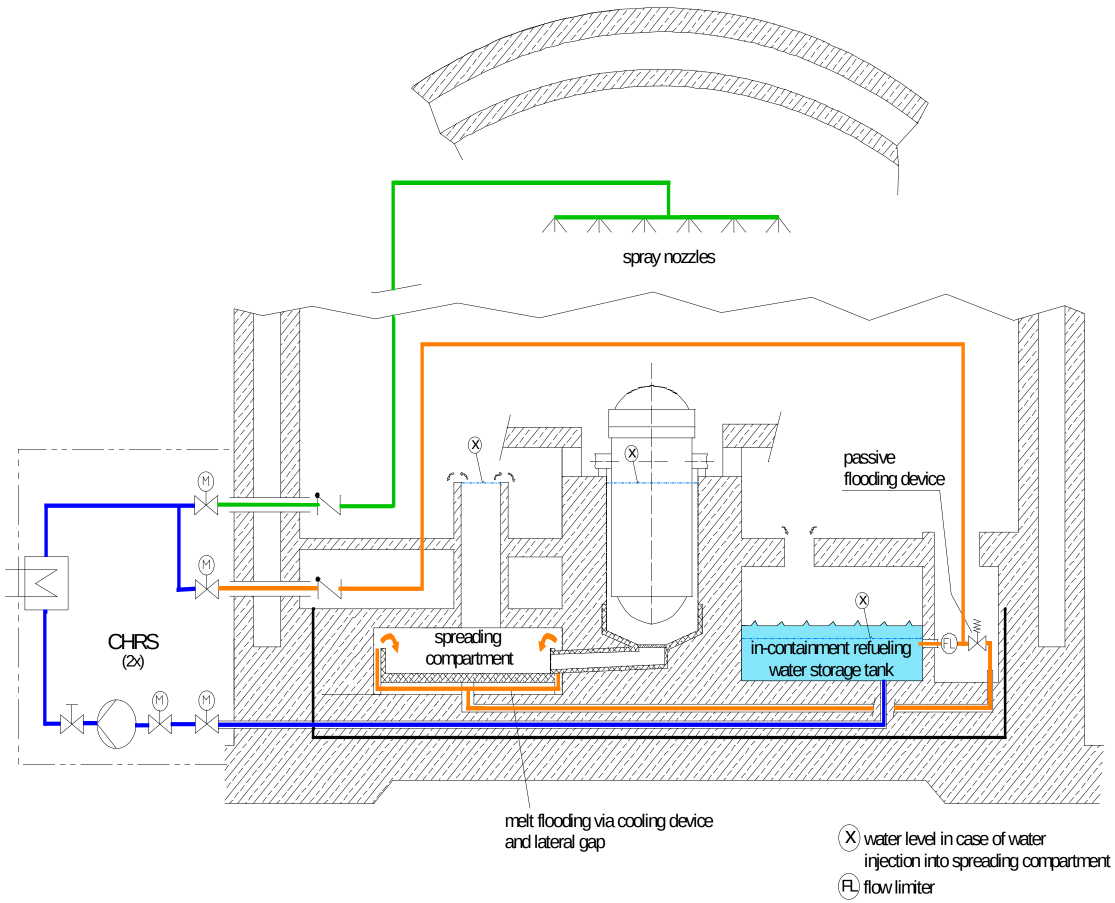
OL3 accident management systems considered in habitability analyses

Emergency core cooling systems

Containment heat removal by sprays

Filtered containment venting system

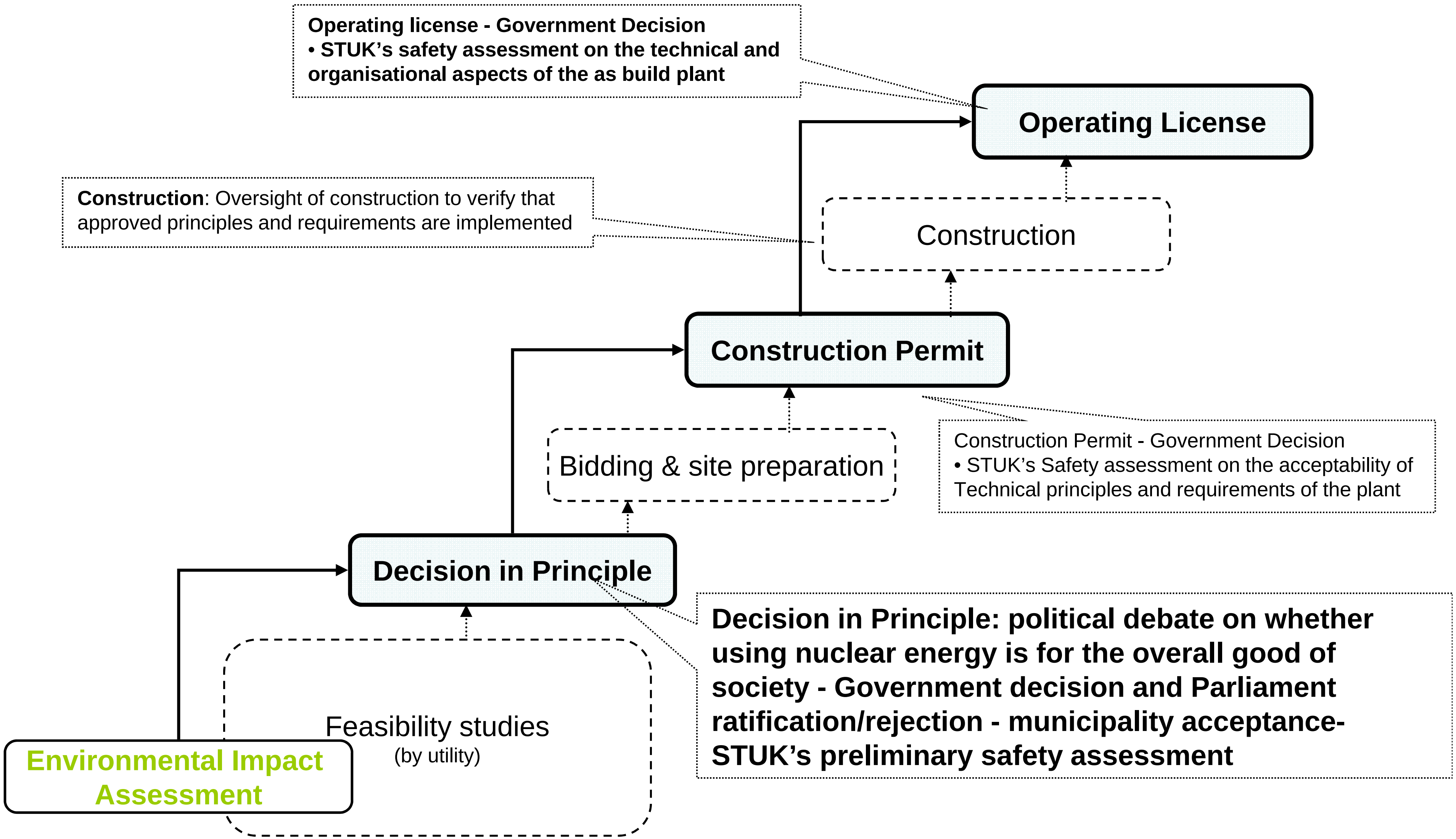
Sampling systems



Regulatory oversight during OL3 construction

- Review of detailed system designs - including radiation safety aspects
- Oversight of construction and manufacturing of components
- Construction Inspection Programme:
  - e.g., project mangement, quality management, safety culture, quality control of construction activities, training of operating personnel, licensee's inspection procedures, use of PRA, **radiation safety requirements**, I&C technology
- Participation in audits made by the licensee
- Topical inspections to the vendor's design process - including radiation safety issues (1st inspection was made already during construction license phase)
- Oversight of commissioning

Licensing stages in Finland based on environmental and nuclear legislation



Fennovoima, new NPP project

- **Site Survey in 2007**
  - **30 areas in 10 municipalities**
  - **detailed assesment of 10 areas**
- **EIA** process and land use planning **2008- 2009**

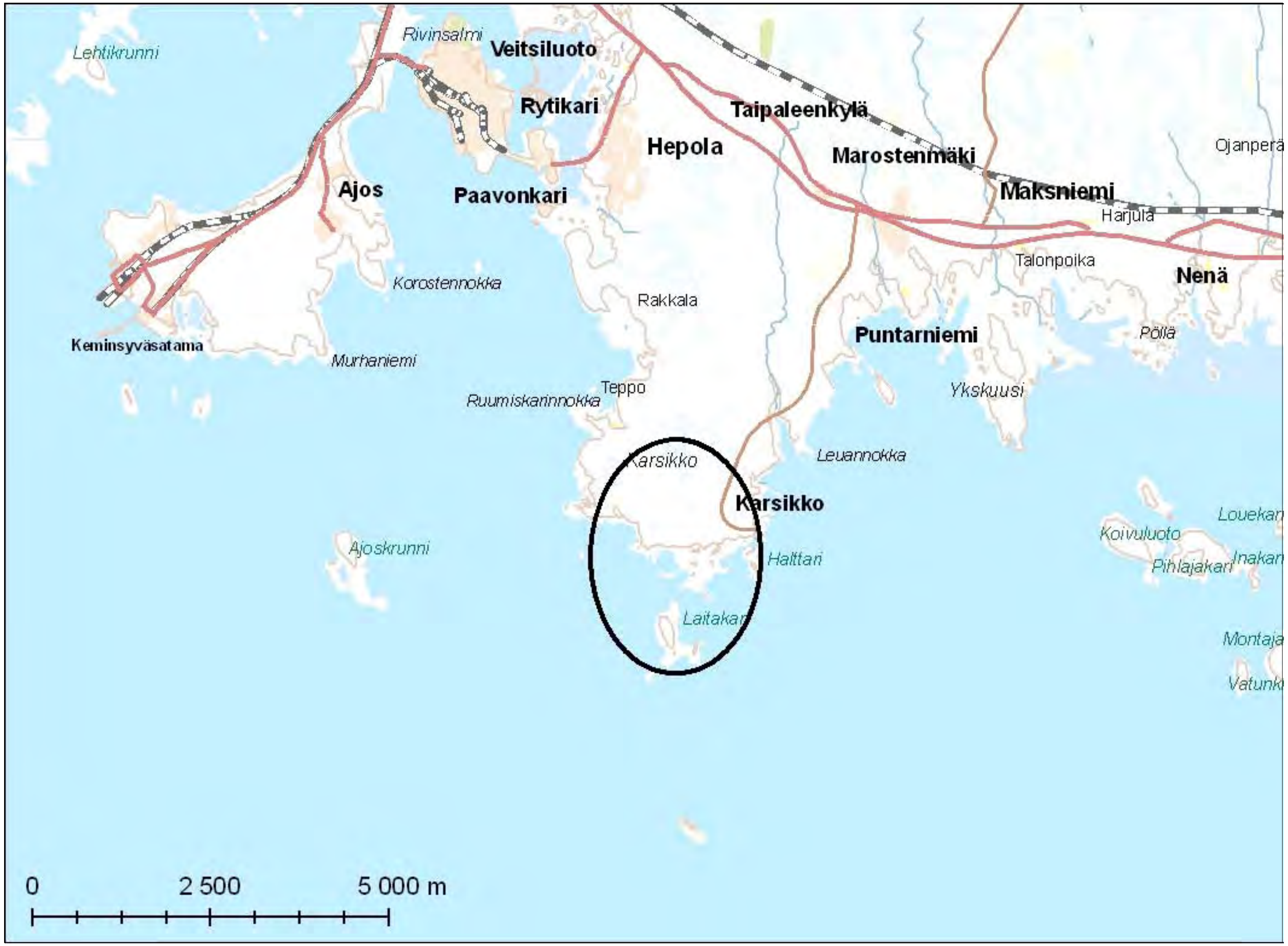
  4 candidate sites
  - **SIMO**, Karsikkoniemi headland
  - **PYHÄJOKI**, Hanhikivi headland
  - **KRISTIINANKAUPUNKI**, Norrskogen
  - **RUOTSINPYHTÄÄ**, Kampuslandet & Gäddbergsön
- NPP project, 1500 - 1800 MW
- Government & Parliament, **2010**

  DiP 2 alternative sites

DiP>

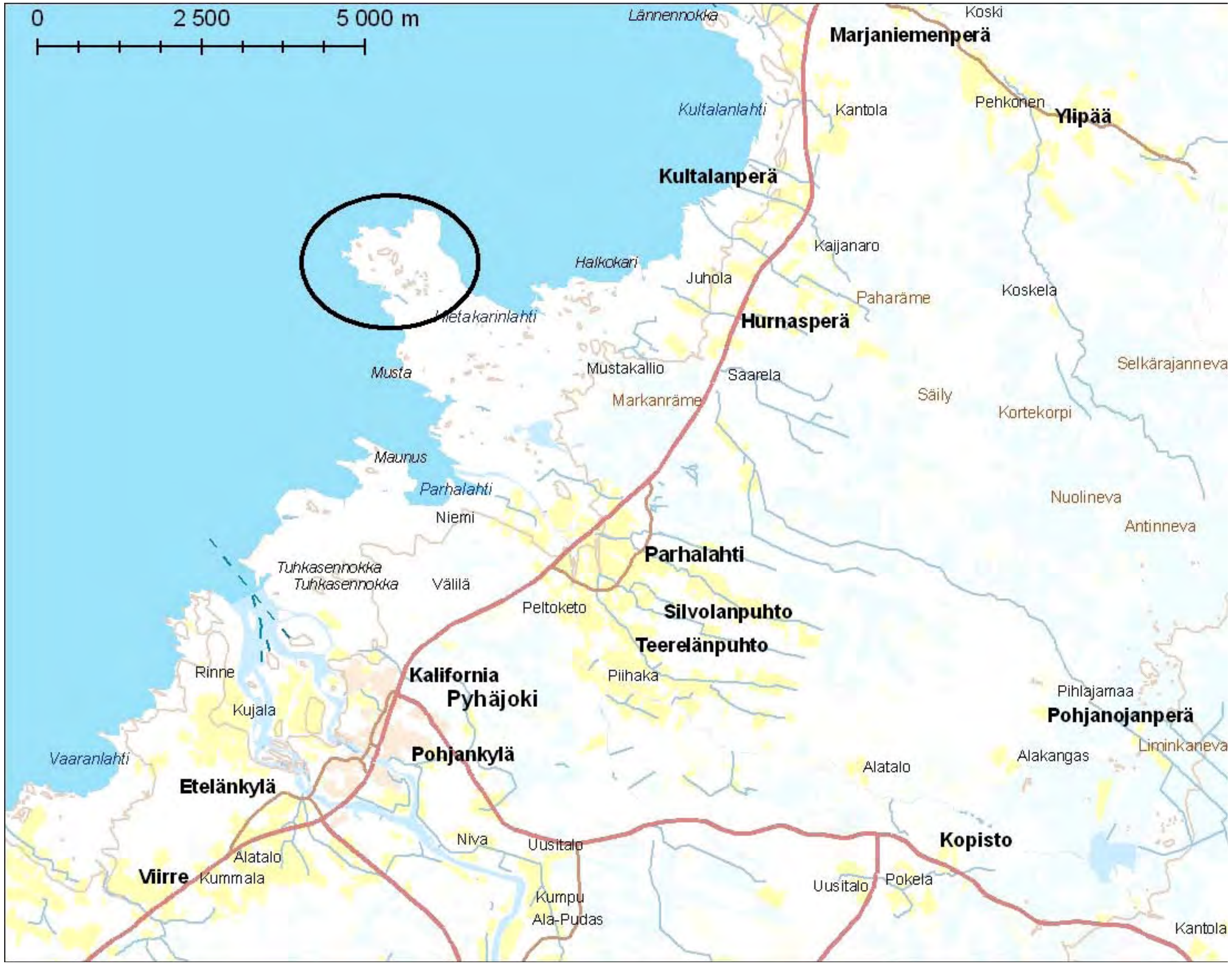
DiP>

Simo, Karsikkoniemi (20 - 30 km from Sweden)



Simo Karsikkoniemi and northern dimension

Pyhäjoki, Hanhikivi (natural environment)



Environmental Impact Assessment (EIA)

IN FINLAND:

- **Act** on Environmental Impact Assessment Procedure (EIA) (**468/1994**) and the Decree on EIA **(713/2006)**

  NPPs, Nuclear Waste disposal facilities, ..

- **Alternative candidate plant sites** may be simultaneously examined during the EIA process and further in the application for a Decision in Principle.

- Finland's **neighbouring countries shall be heard** where deemed necessary by virtue of the Convention on Environmental Impact Assessment in a Transboundary Context

Parallel also a **hearing of land use plans** (Espoo Convention)

EIA programme and report (main issues)

- Effects of construction
- **Land-use and landscape**
- Effects on the **sea water and fishing**
- Noise etc
- **Waste** and their effects
- Effects on ground and bedrock and groundwater
- Effects on **plants, animals and nature protection areas**
- **Traffic**
- **Electric power lines**
- **Social effects**

EIA report, certain nuclear and radiation issues

- **Project plan**
- **Radiation and health**
- **Radioactive releases, normal conditions**
- **Accident consequences and mitigation**
  The scope of to be considered is not exactly defined in legislation
  Based on expert discussions, a release corresponding to the severe accident limit 100 % Noble gases & 1000 TBq I-131 & 100 TBq Cs 137 was analysed and presented in the EIA reports
- **Nuclear fuel cycle (waste disposal plan)**
- **Decommissioning**

EIA IS EARLY PROCESS. INFORMATION THUS VERY GENERAL

Fennovoima NPP & Northmost Site - Simo
DiP, STUK statement

- **Preliminary safety assesment**
  - Can be planned and constructed to fullfill Finnish regulations
- **Feasibility of the site**
  - **Northern conditions (meteo, see)**

    > plant design
  - **Seismic data & requirements**

    > plant design
  - **Air traffic (K-T airport)**

    > plant design and air control
  - **Preparedness for emergencies**

    > some new planning and tools
- **Fennovoima resources and skills**

- **Plant alternatives**
  - Areva **EPR**
  - Areva **KERENA**
  - Toshiba-Westinghouse **ABWR**

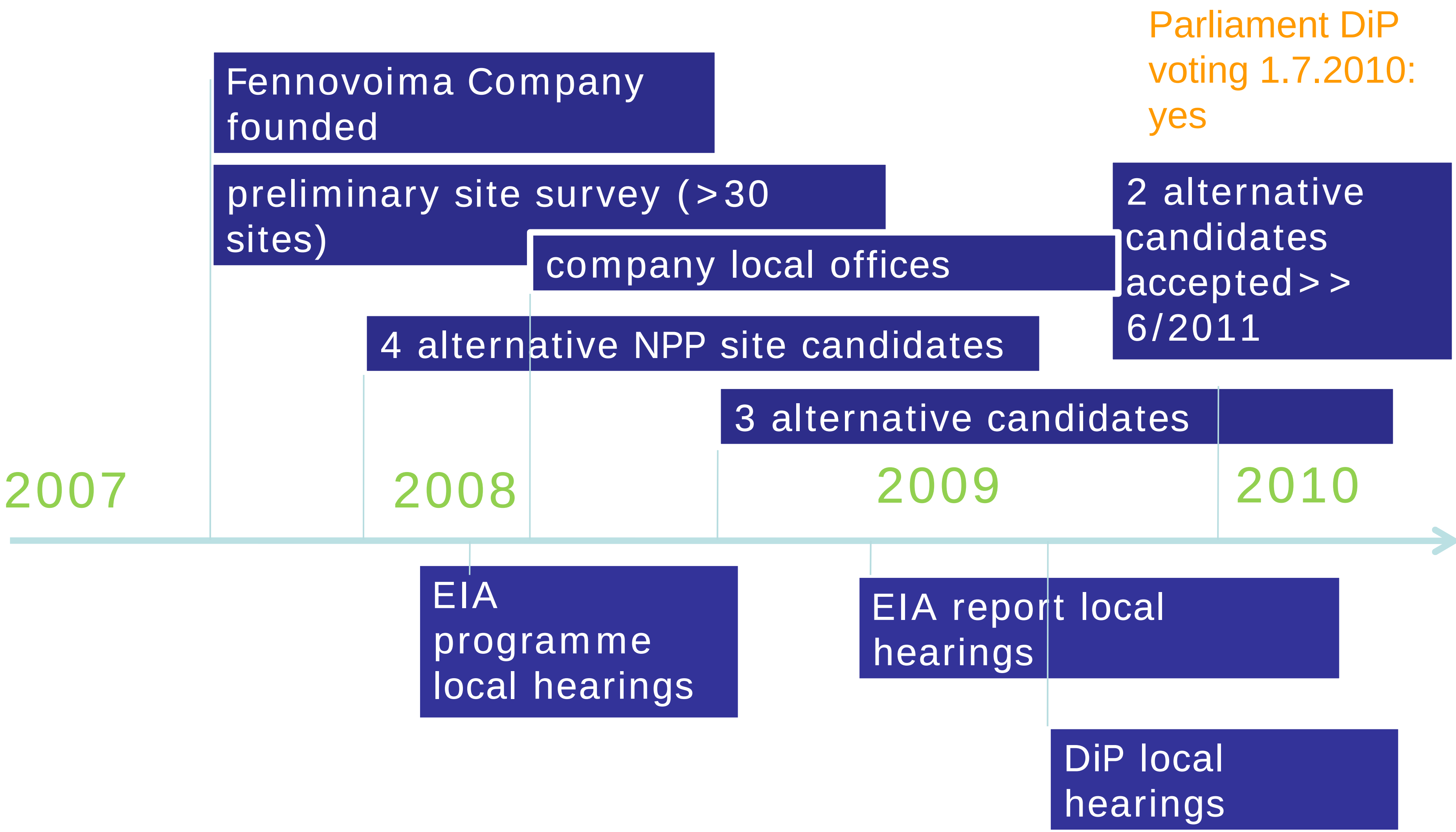
STUK contact to Stakeholders

- participation in official hearings on site (EIA & DiP)
- lectures invited by the municipalities concerned, and local education centres
- participation in critical discussion panels (with Greenpeace, Swedish Actors, local and national action groups)
- interviews to media (morning papers, radio, TV)

open web, questions/answers

Issues questioned:

- normal life and employment; agriculture (potatoes, tomates..), fishing, see water quality
- emergency planning zones, housing and traffic
- nuclear waste management


Parliament DiP voting 1.7.2010: yes
Fennovoima Company founded
preliminary site survey (>30 sites)
company local offices
2 alternative candidates accepted>> 6/2011
4 alternative NPP site candidates
3 alternative candidates
2007
2008
2009
2010
EIA programme local hearings
EIA report local hearings
DiP local hearings


Simo Karsikkoniemi, possibly a new NPP operating site in 2020
Fishermen in winter

Thank you for attention !

More information available:

www.stuk.fi (also in Swedish and English), e.g.:

- STUK's YVL Guides
- STUK's safety assessment concerning Olkiluoto unit 3 construction licence
- STUK's preliminary safety assessments concerning the possible new plant units OL 4, FV and LO 3 in Finland

OR: Contact stuk@stuk.fi,
olli.vilkamo@stuk.fi GSM + 358 400 817981

François Barré – Olivier Marchand – Yann Monerie – Frédéric Perales

Institut de Radioprotection et de Sûreté Nucléaire (IRSN)

Advanced modelling of complex material properties – a significant step toward predictive calculation of clad ruptures

Summary of the presentation

- Industrial context
- Safety organisation needs
- Advances R&D modelling
- Numerical platform: Xper
- Applications
- Conclusion & Outlooks

The Industrial context

- Economic context, Competitiveness, Flexibility
- Continuous evolution of fuel technologies
  - Claddings: Zr2, Zr4, Zirlo, O-Zirlo, M5, MMDA .... Quaternary alloys
  - Fuel pellets: Doped UO2, Gd fuel, Short pellets, doped MOX, ...
  - Assemblies
    - ☞ High Performance Fuel
- Evolution in-NPP fuel management
  - Increase of reactor power, burn-up, ...
  - Flexibility of operating modes, ...

⇨ Continuous increase of mechanical, chemical and thermal stresses

⇨ Regularly appearance of new fuel products

The Safety Organisation needs (1/2)

- **Fuel is a complex object:**
  - In depth knowledge of coupled phenomena are necessary
- **Very large number of fuel rods under operation in NPP** (defect probability)
  - E.g. in France: 10 000 assemblies: 3 millions of fuel rods
- **Diversity in fuel suppliers + fast evolution of technologies**
- **Diversity in operating conditions**

⇨ In incident or accident conditions,

large number of initial conditions and responses to stresses

to be taken into account in safety assessment

The Safety Organisation needs (2/2)

- **Safety objectives**
  - To guarantee a proper behaviour of the first barrier (normal operation, incident and accident conditions (LOCA, RIA, …))
- **Expert assessment of fuels** (defect probability)
  - Complexity of the Physics
  - Large variety of technologies and operating conditions
  - Diversity of possible incident and accident
- **Safety Assessment should be based on a well thought simulation / experimentation R&D strategy**
- **TSOs should be leader in R&D investigation on support of safety Assessment**

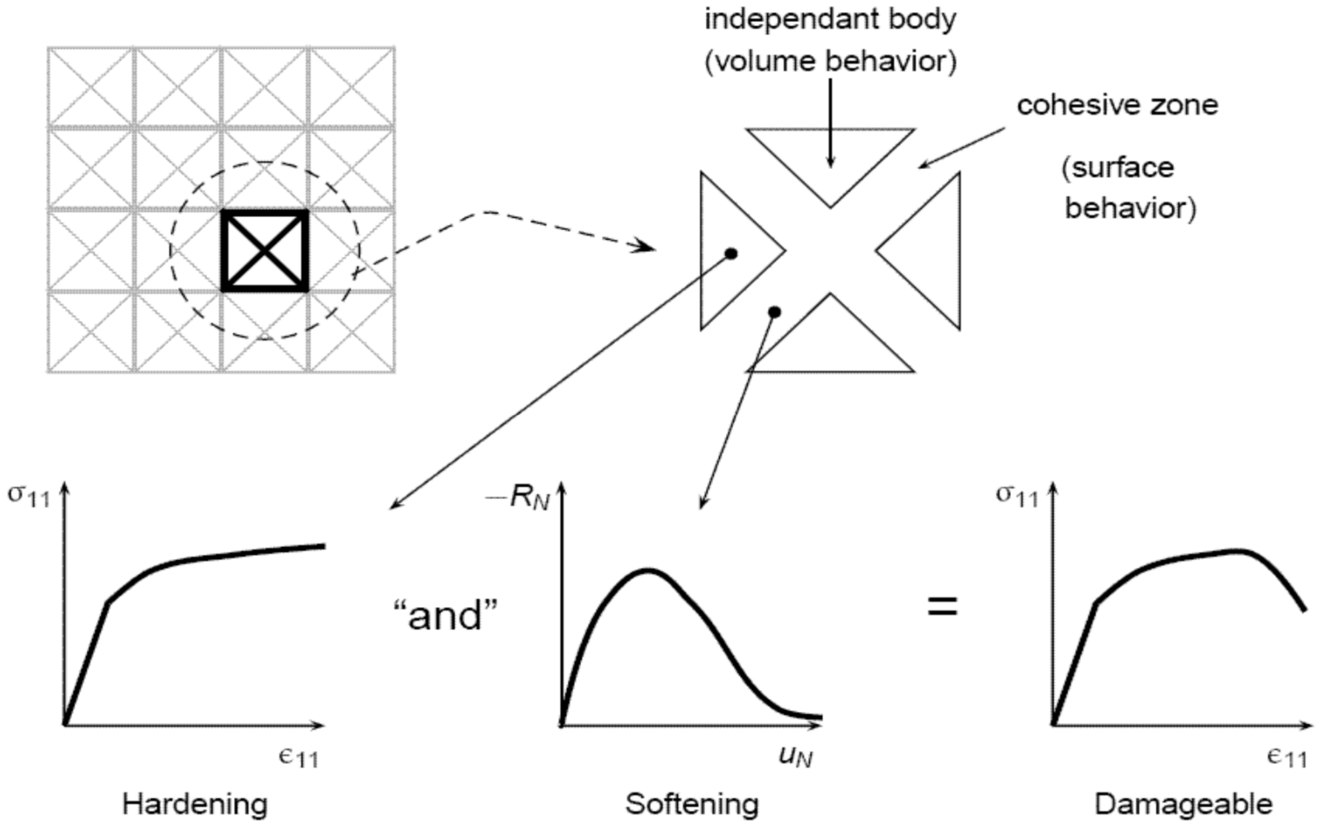
→ IRSN R&D strategy on R.I.A and L.O.C.A.

Advanced R&D

- IRSN R&D strategy
  - an advanced R&D, in order to understand more fully physical phenomena at small scales,
  - In-pile and Out-of-Pile experiments,
  - an applied R&D, in order to be able to establish physical basis for the Safety Analysis
- IRSN advanced R&D
  - development of advanced modelling to predict the behaviour of materials in all situations
  - the synthesis and enhancement of the knowledge acquired in nuclear safety support software (SCANAIR – DRACCAR)

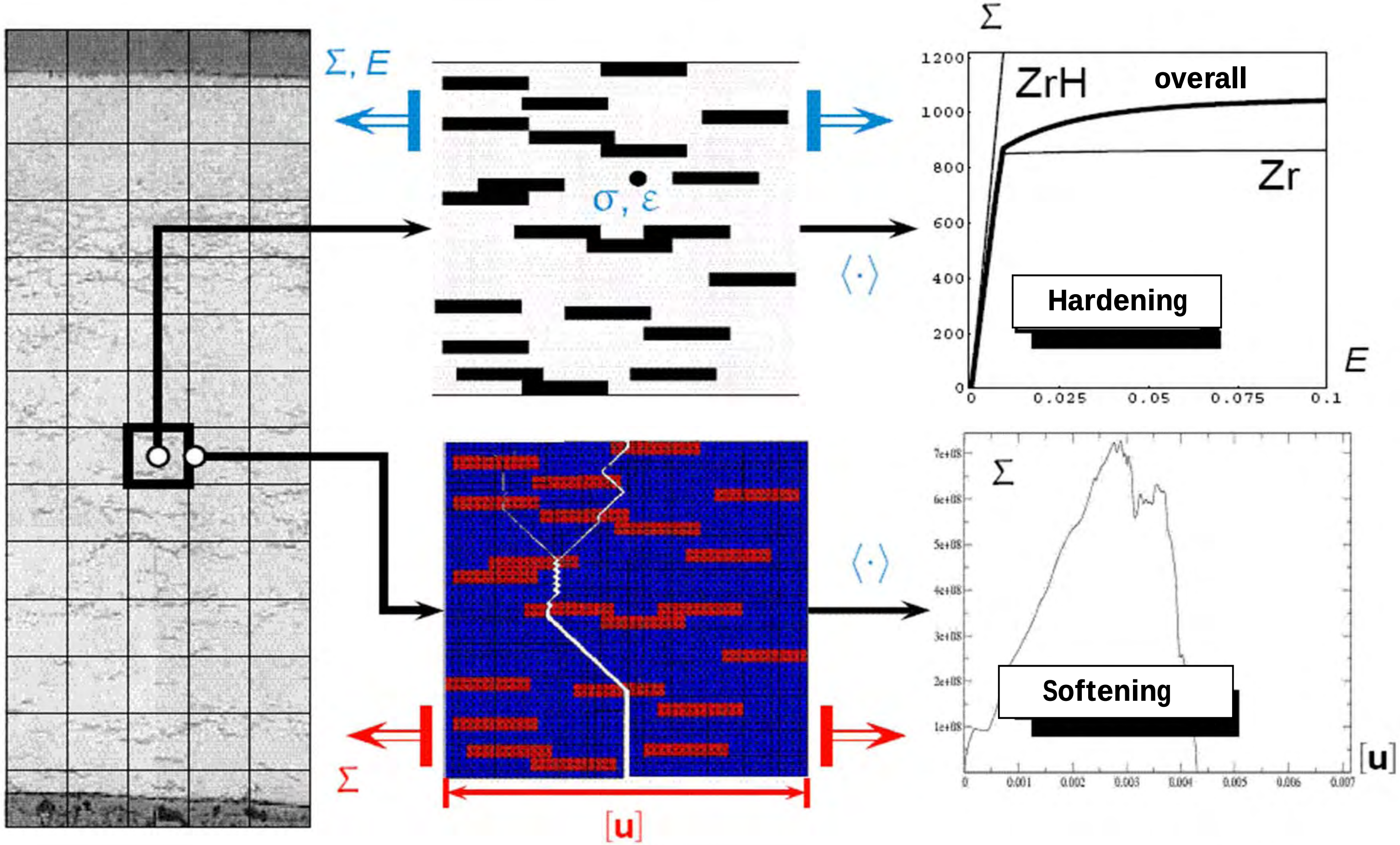
Advanced R&D modelling: fracture

- Multibody approach



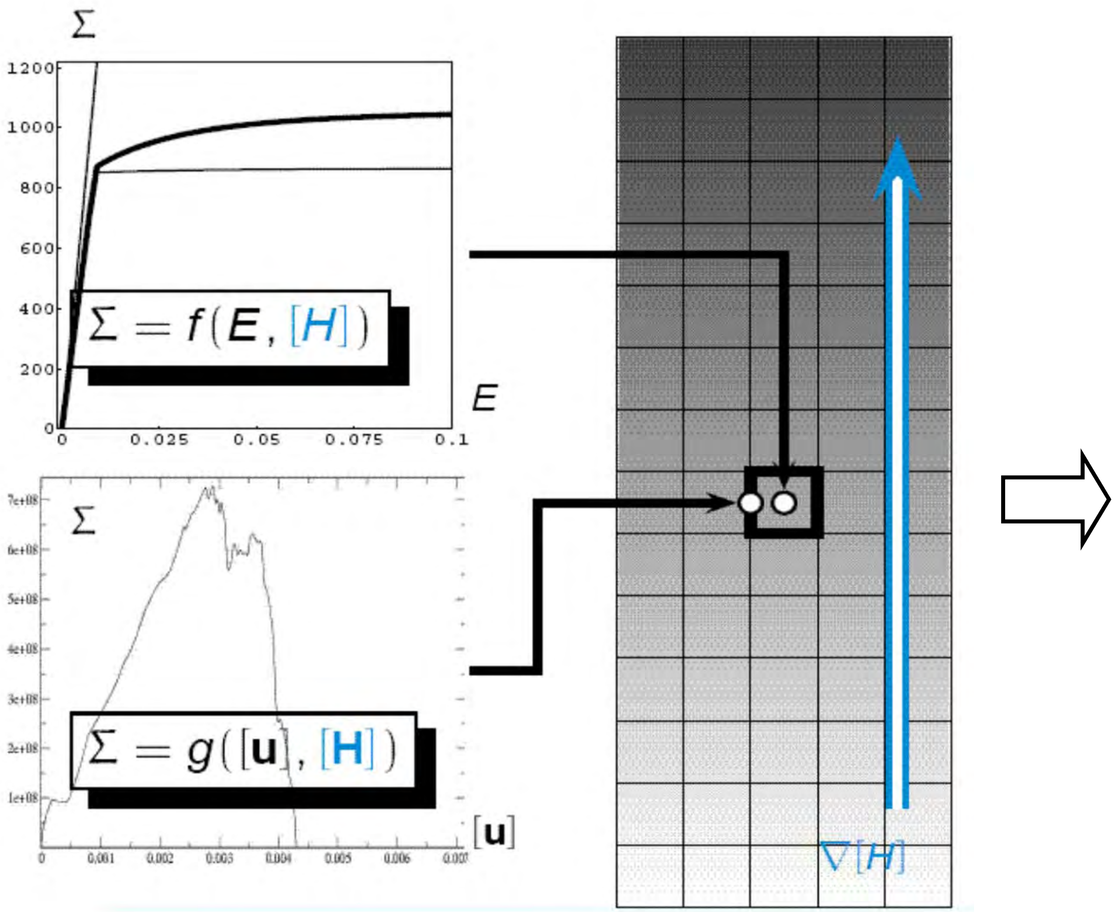
Advanced R&D modelling: multiscale (1/2)

- Microscale



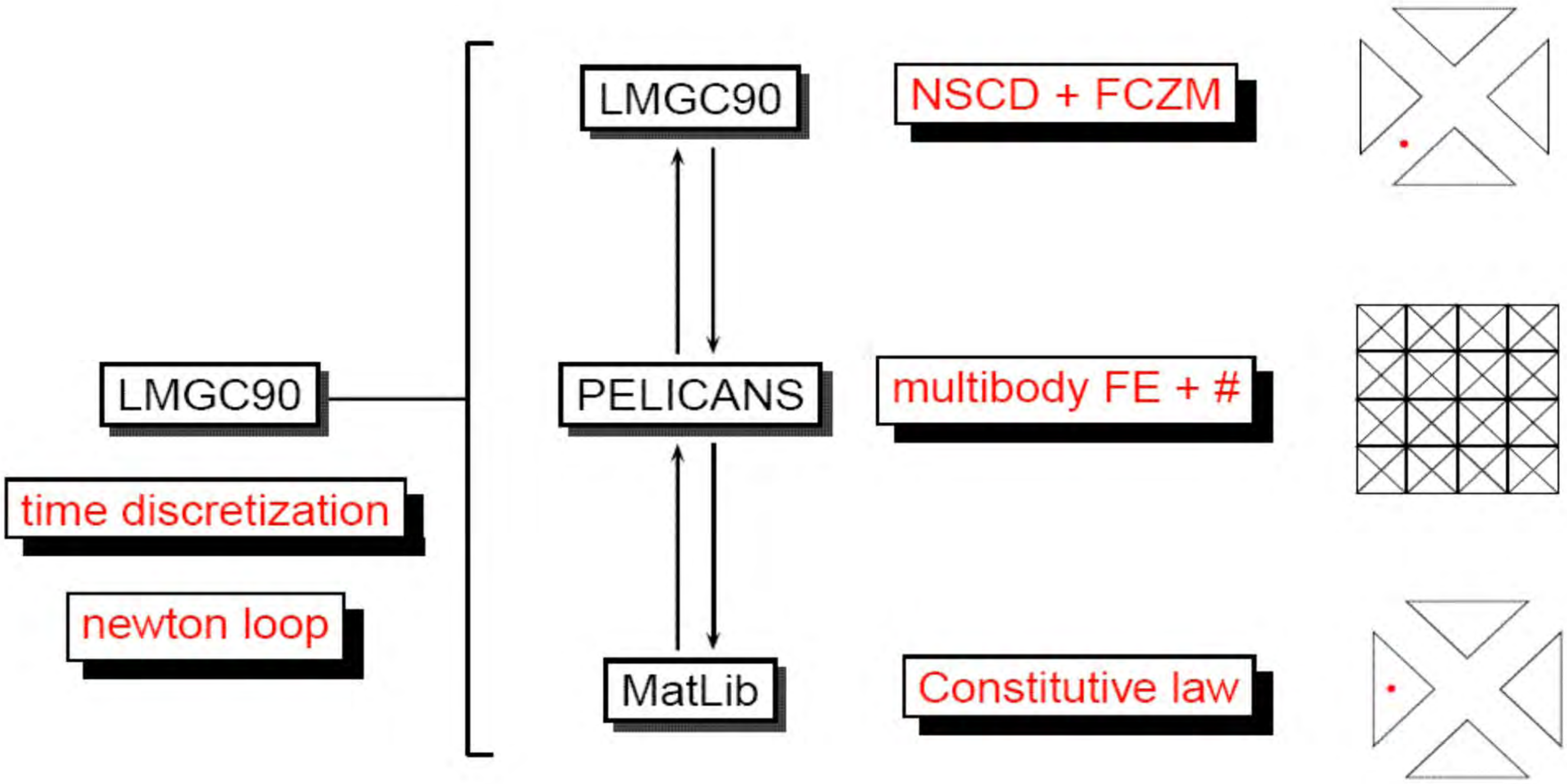
Advanced R&D modelling: multiscale (2/2)

- Macroscale



Overall behavior:
Elastoplastic
Damageable
(multifissuration)

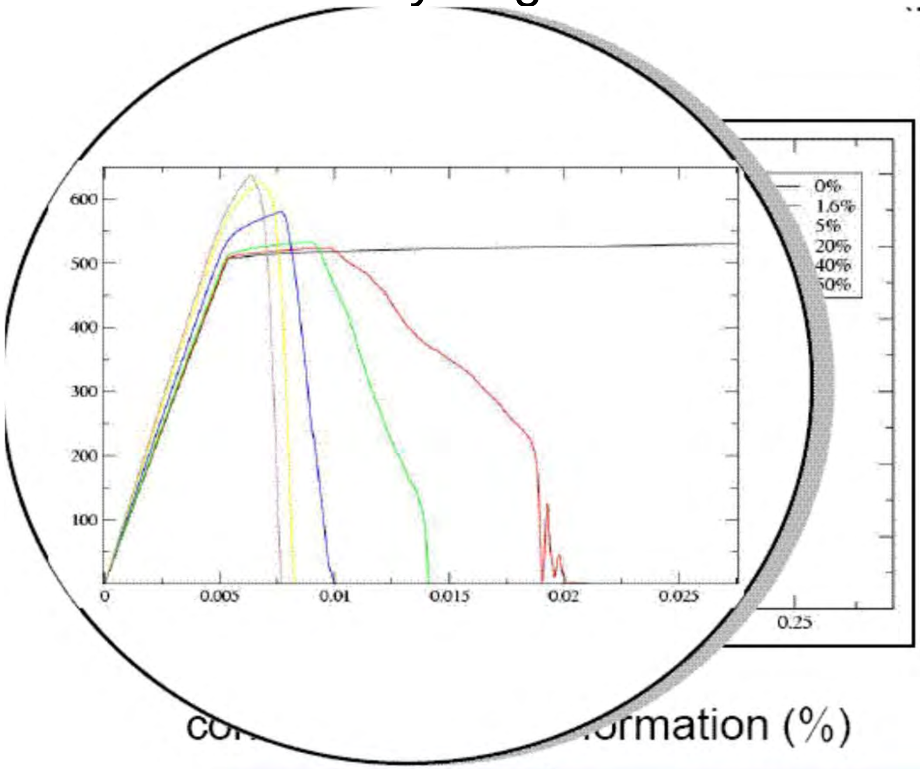
Numerical platform: Xper

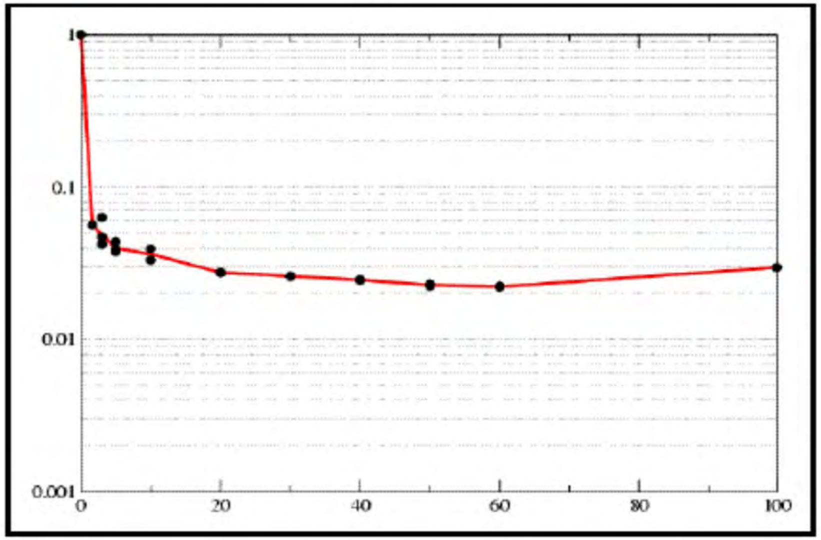


New hydrided Zircaloy law for SCANAIR code

- SCANAIR = RIA transient fuel rod code
- Several hydrogen content

s (various samples)



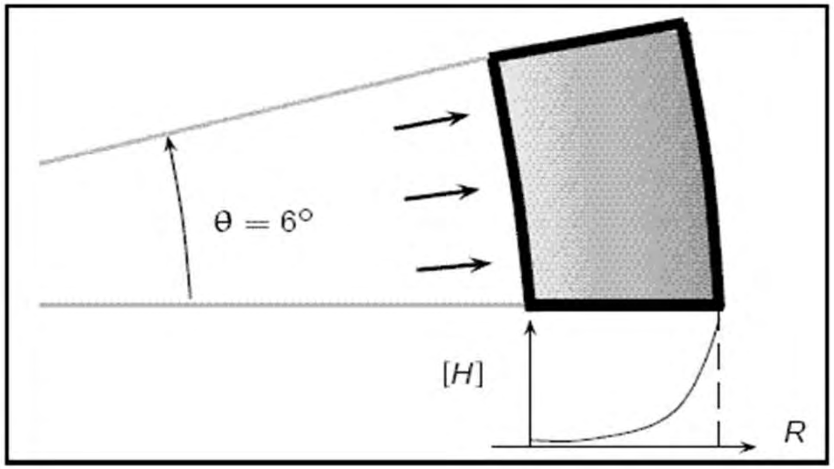


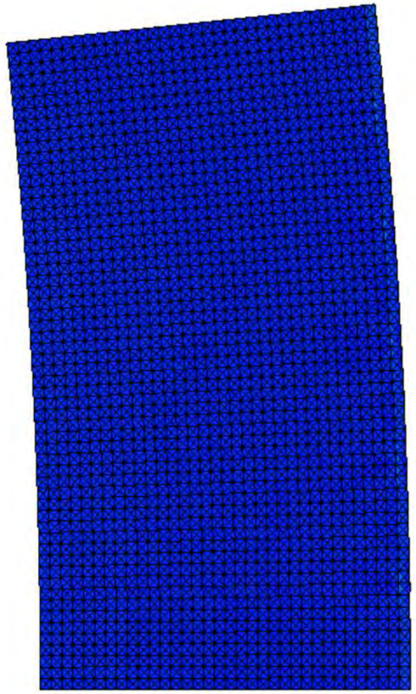
Energy vs [H]

Implementation of the law in the SCANAIR software in progress

RIA rupture simulation

- Part of a fuel rod
- Functionnaly graded material (∇ [H])
- Elastoplastic hardening material
- Dynamic loading: $\dot{\bar{F}}_r > 10s^{-1}$

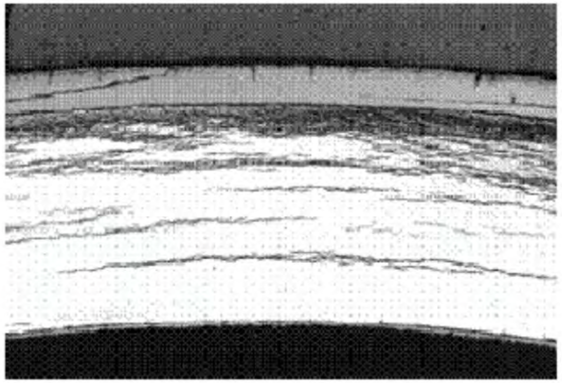




Understanding of the RIA rupture mechanims

RIA tests interpretation (1/3)

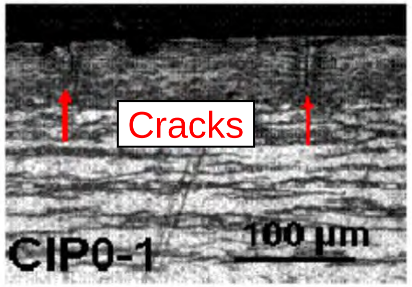
CIP0-1, VA-1 and VA-3 before test (20°C)

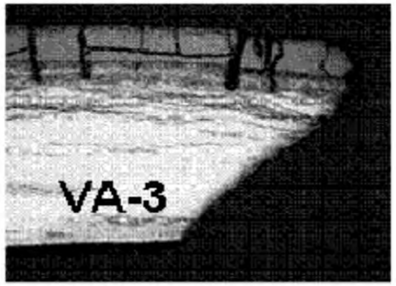


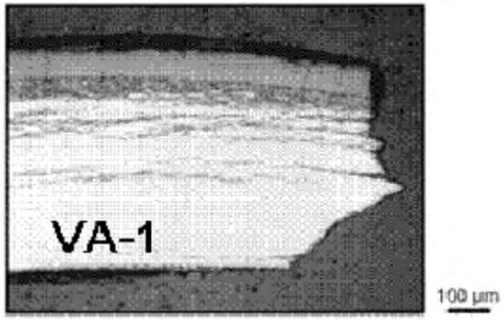
280°C

After test

20°C







No Failure

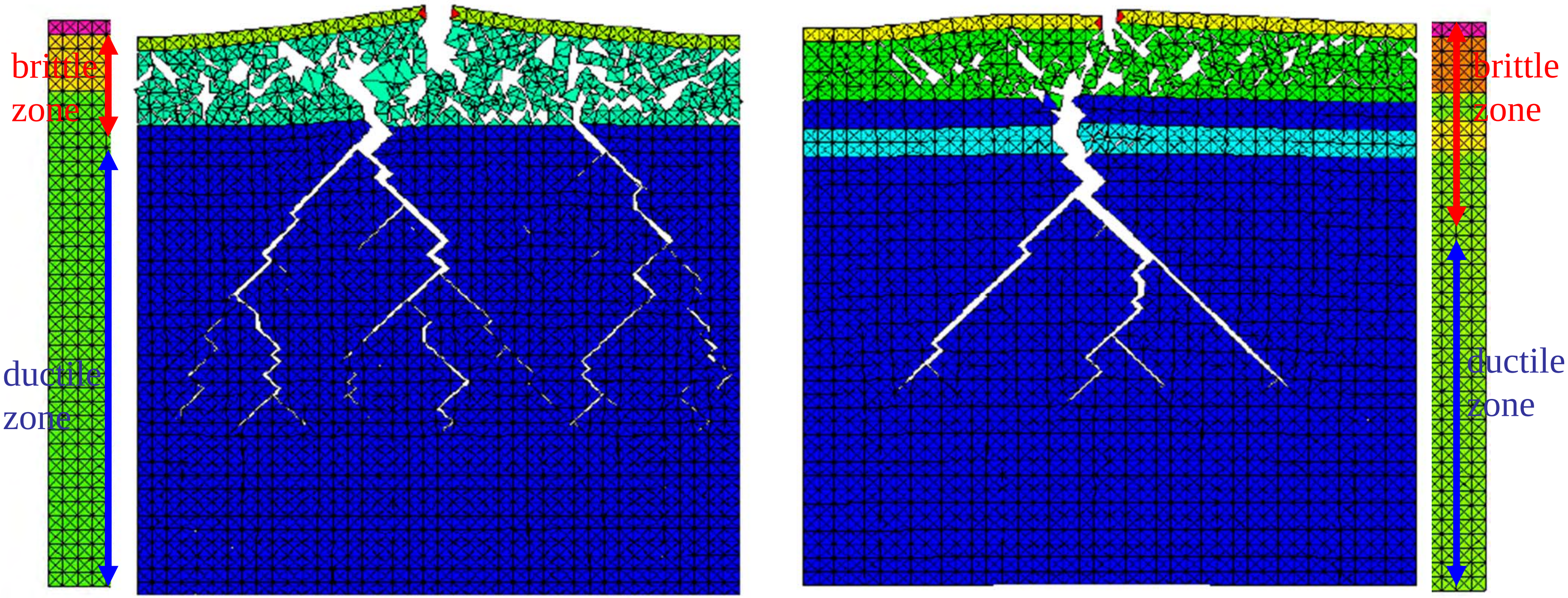
Failure

Failure

Needs advanced R&D to understand

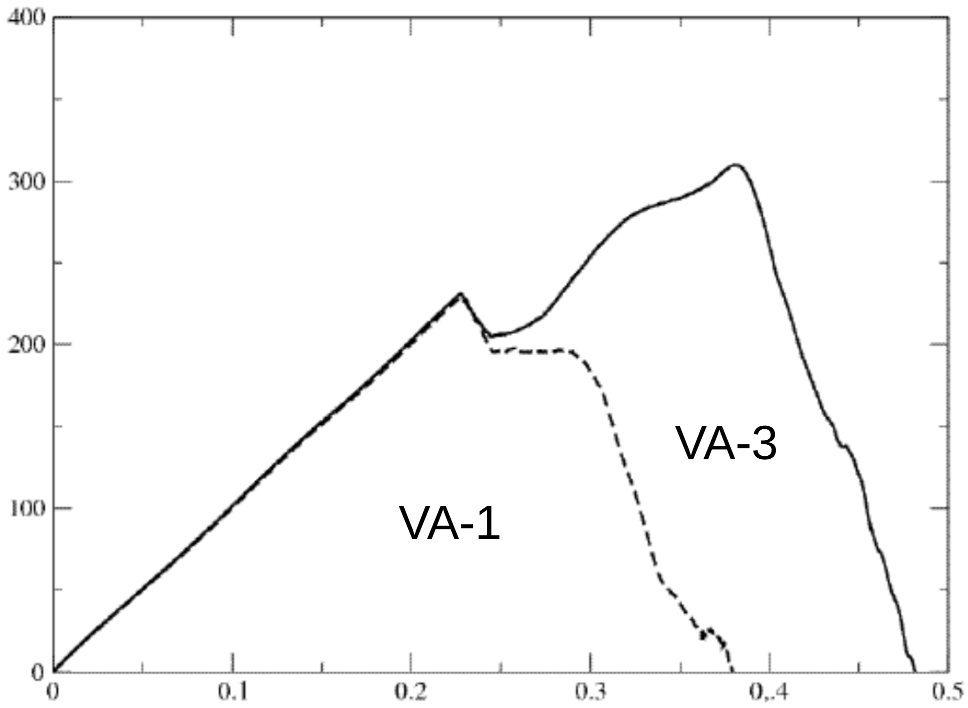
RIA tests interpretation (2/3)

- Tests modelling :
  - VA-3: Matrix (1% H) + Rim (60% H)
  - VA-1: Matrix (Zr) + Rim (60% H) + Rim (20% H)



RIA tests interpretation (3/3)

- 2 hydride rims more deleterious than 1 hydride rim



Stress vs Strain

Understanding of the rupture mechanims of tests VA-1 and VA-3

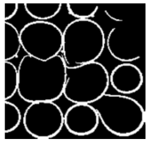
Qualification of DRACCAR software (1/3)

- DRACCAR = LOCA transient on nuclear power plant code

Ballooning of claddings

↓

- Mechanical interactions between rods



PHEBUS LOCA TEST

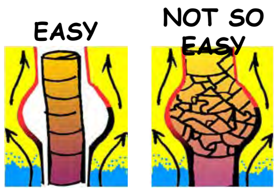
↓

- Burst criteria

- Fuel Relocation

↓

- Coolability around ballooned regions ?



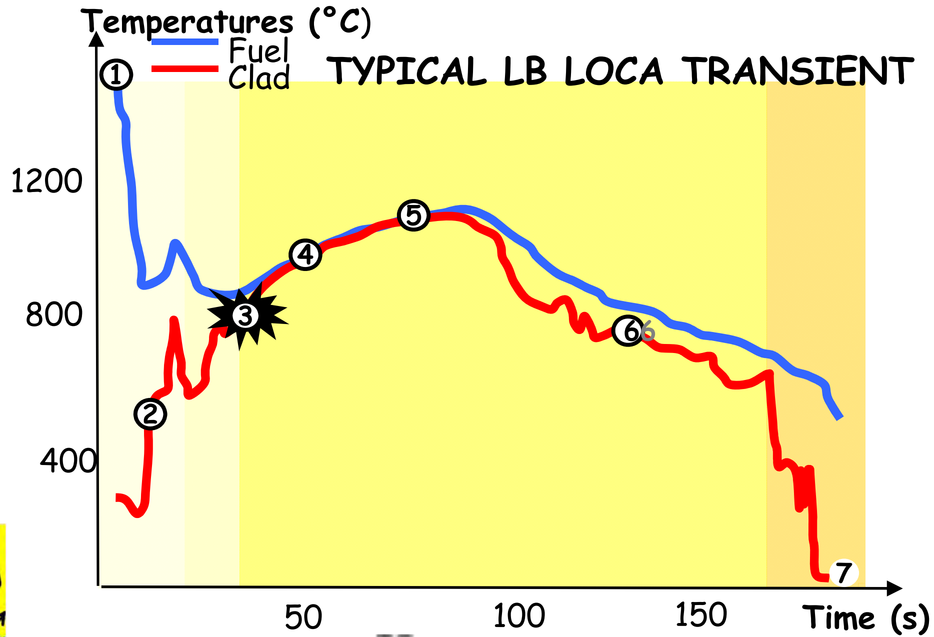
- Reflooding ?

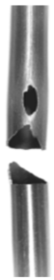
↓

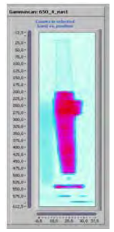
- Clads Oxidation / hydriring

→ Rupture / Coolability ?

- Residual ductility

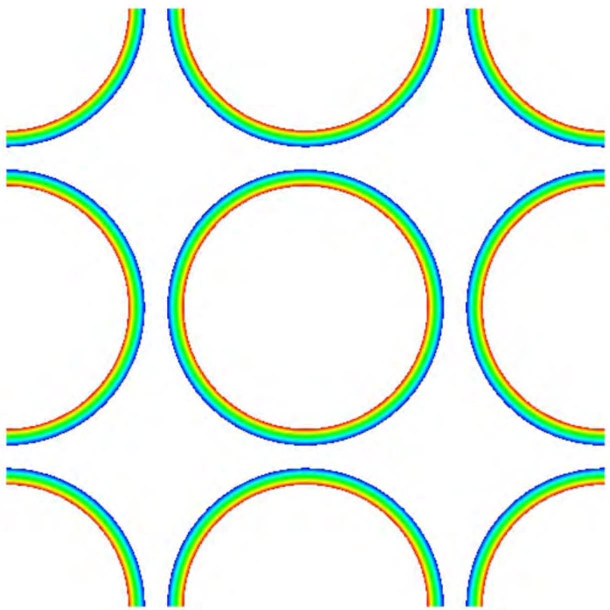






Qualification of DRACCAR software (2/3)

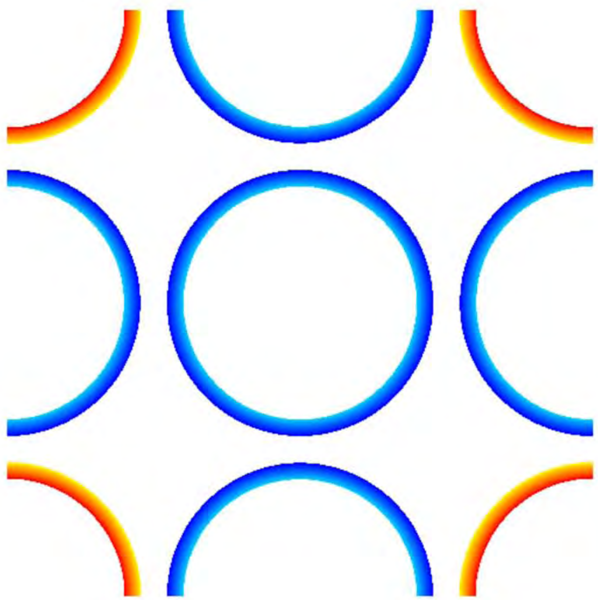
- 4 rods
- Symetric loading: same internal pressure



Integral criteria, such as flow blockage ratio, well evaluated

Qualification of DRACCAR software (3/3)

- 4 rods
- Nonsymetric loading: different internal pressure



local fields defaulted by the Xper simulation

Implementation of a new contact law in progress

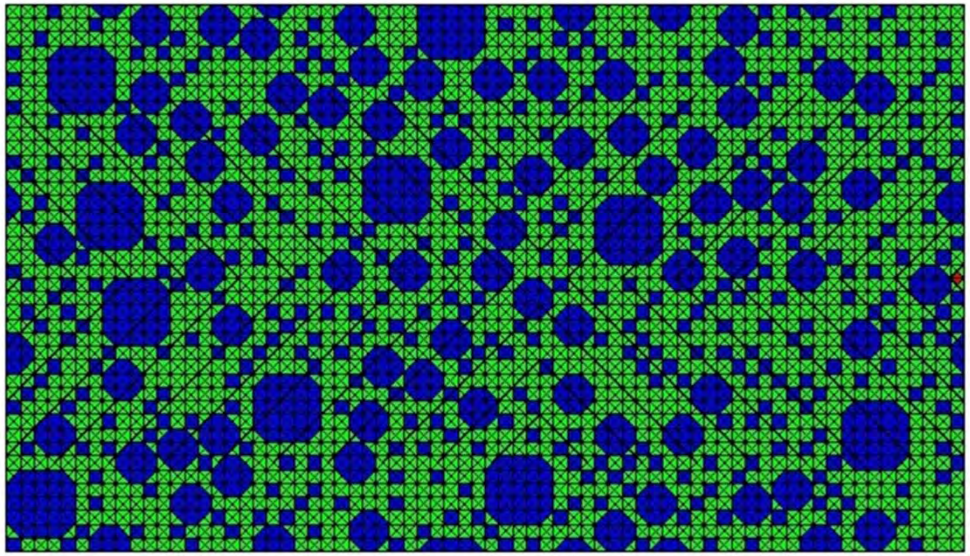
Conclusion

- To be in a position to validate products, assessment methodologies and operating conditions, IRSN is proposing R&D programmes, in the framework of international collaboration, in particular with TSOs

- In this paper we focused on the advanced R&D programme proposed by IRSN in the context of a revision of acceptance criteria and the evolution of methodologies used for the safety assessment for LOCA and RIA.

Outlooks

- Generic modelling
- Applications to the topic « ageing of nuclear power plants »
  - Metals (reactor vessel)
  - Concretes (nuclear power plant concrete structures)
  - …



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Towards a German Safety Case - The ISIBEL Project

The ISIBEL Project

Yardstick

- ability to develop a safety case according to state-of-the-art

Objectives

- refinement of safety concept for HLW disposal in rock salt
- development of a novel approach to demonstrate safety
- identification of necessary R&D

Project partners

- Bundesanstalt für Geowissenschaften und Rohstoffe, Hannover
- DBE Technology, Peine
- Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH

Research project

- funded by German Ministry of Economics and Technology (BMWi)
- duration: October 2005 – May 2012

Fundamentals

Geology

- available knowledge concerning salt domes in Northern Germany (stylised site model)

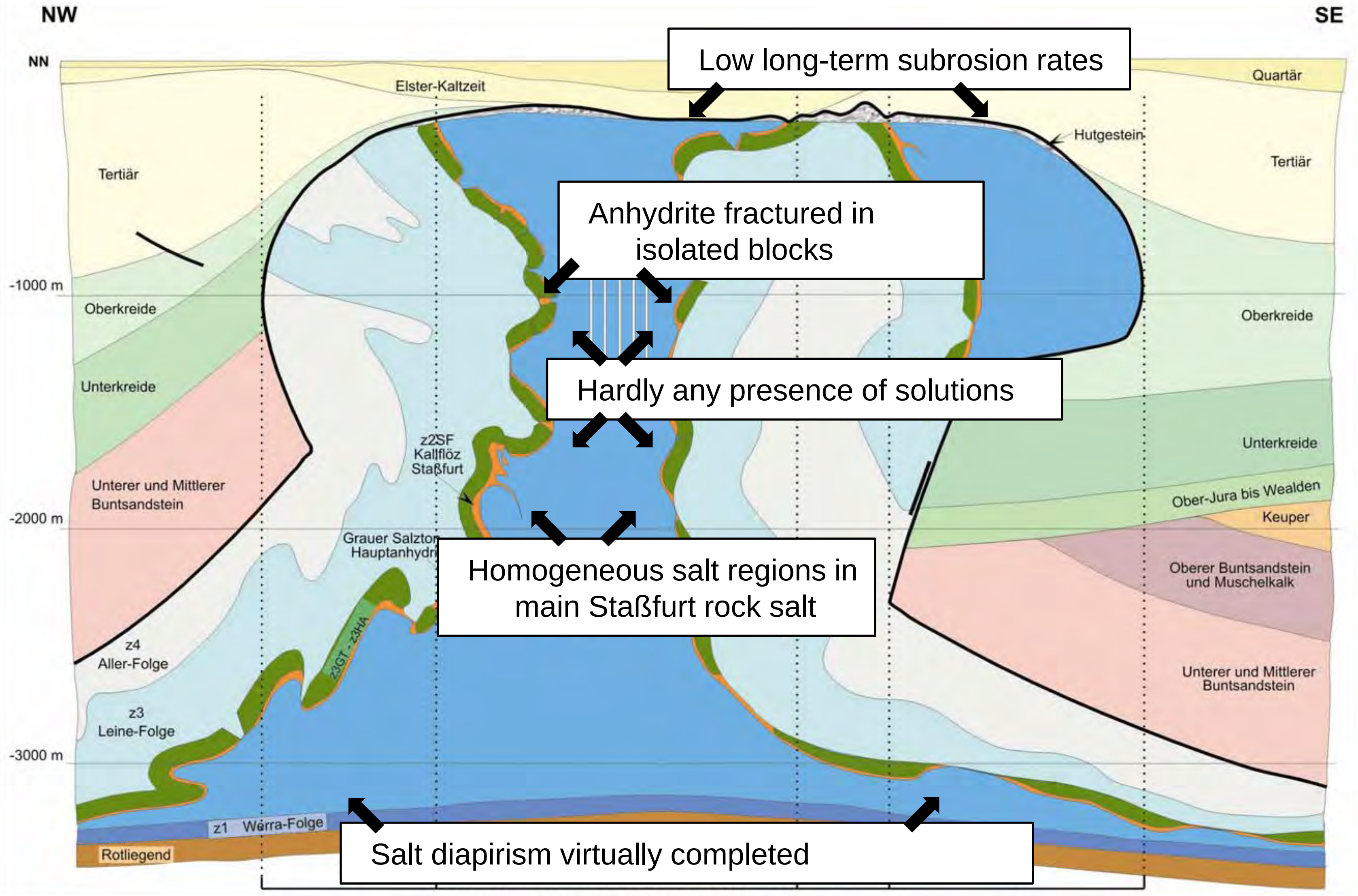
Waste amount and characteristics

- amount according to present status of German Atomic Energy Act (phase-out decision)
  - spent fuel
    - 2,045 POLLUX casks or 6,817 BSK3 casks
  - waste from reprocessing (vitrified or compacted)
    - 3,767 CSD-V
    - 6,902 CSD-C + 560 CSD-B

Repository concept

- based on reference repository concept of 1998
- including new waste types (BSK3) and more detailed design of geotechnical barriers

Stylised Site Model - Important Features



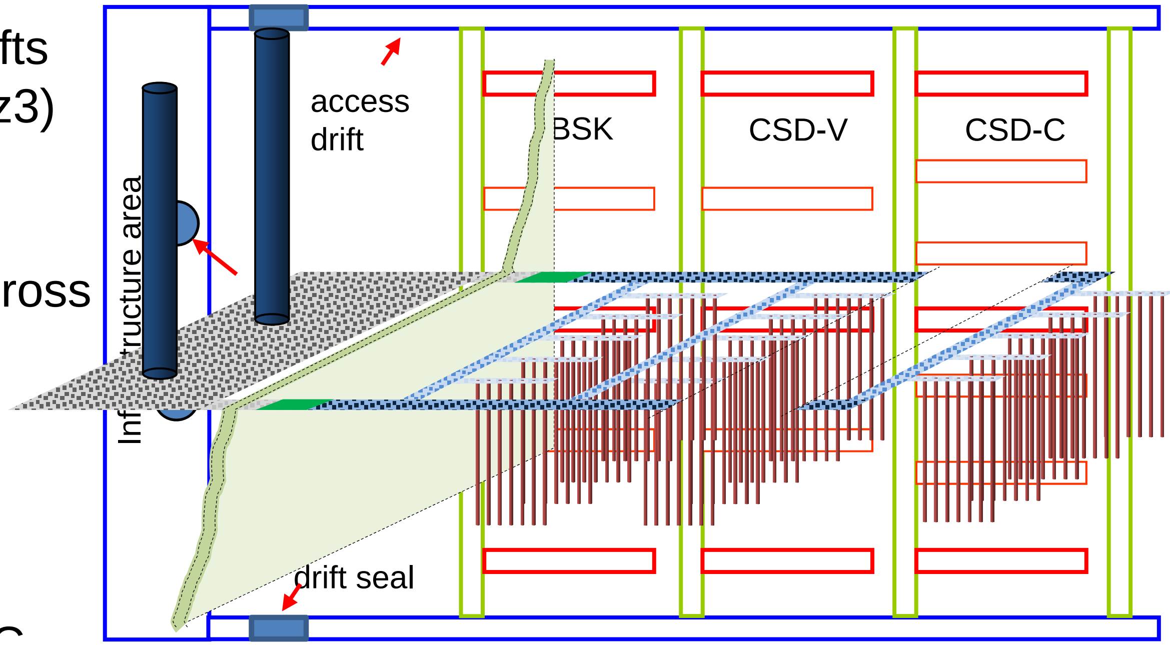
Schematic Repository Layout

Infrastructure area wih two shafts

- located in Leine salt facies (z3)

Two access drifts embracing disposal areas, connected by cross cuts

- located in Staßfurt main rock salt (z2HS)
- several disposal areas
  – 8 BSK3, 1 CSD-V, 1 CSD-C



Designed high-performance shaft and drift seals

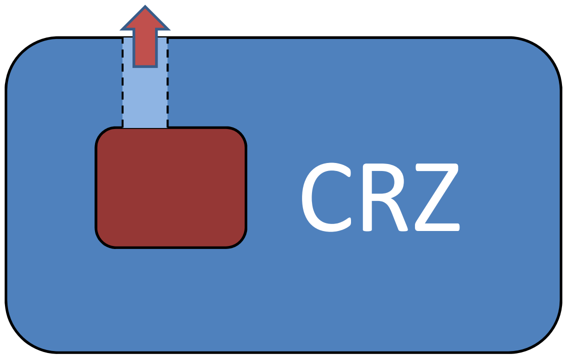
Repository void volumes is backfilled with crushed salt

Repository void volumes will fade away

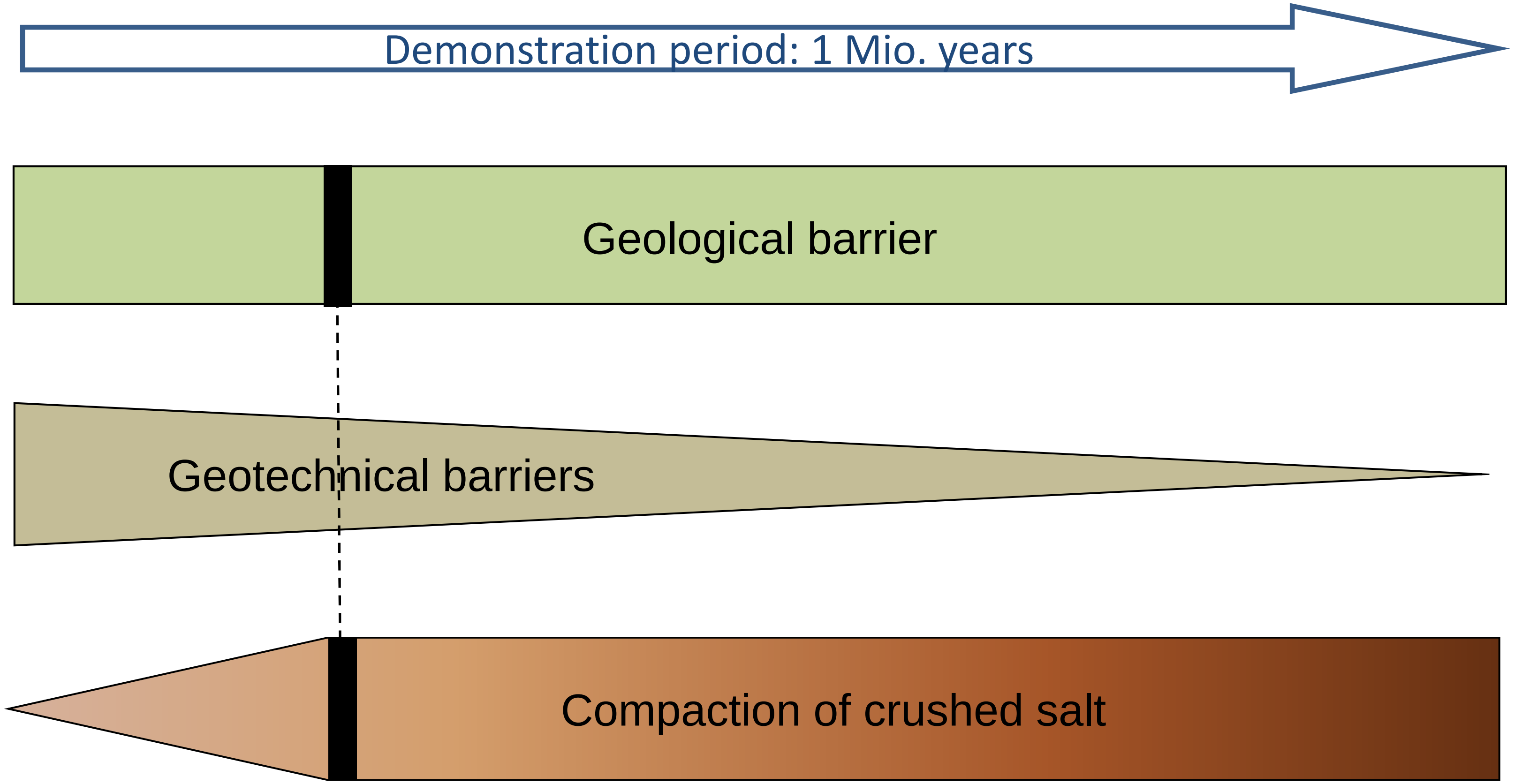
The Safety Concept

Safety concept focusses on safe containment

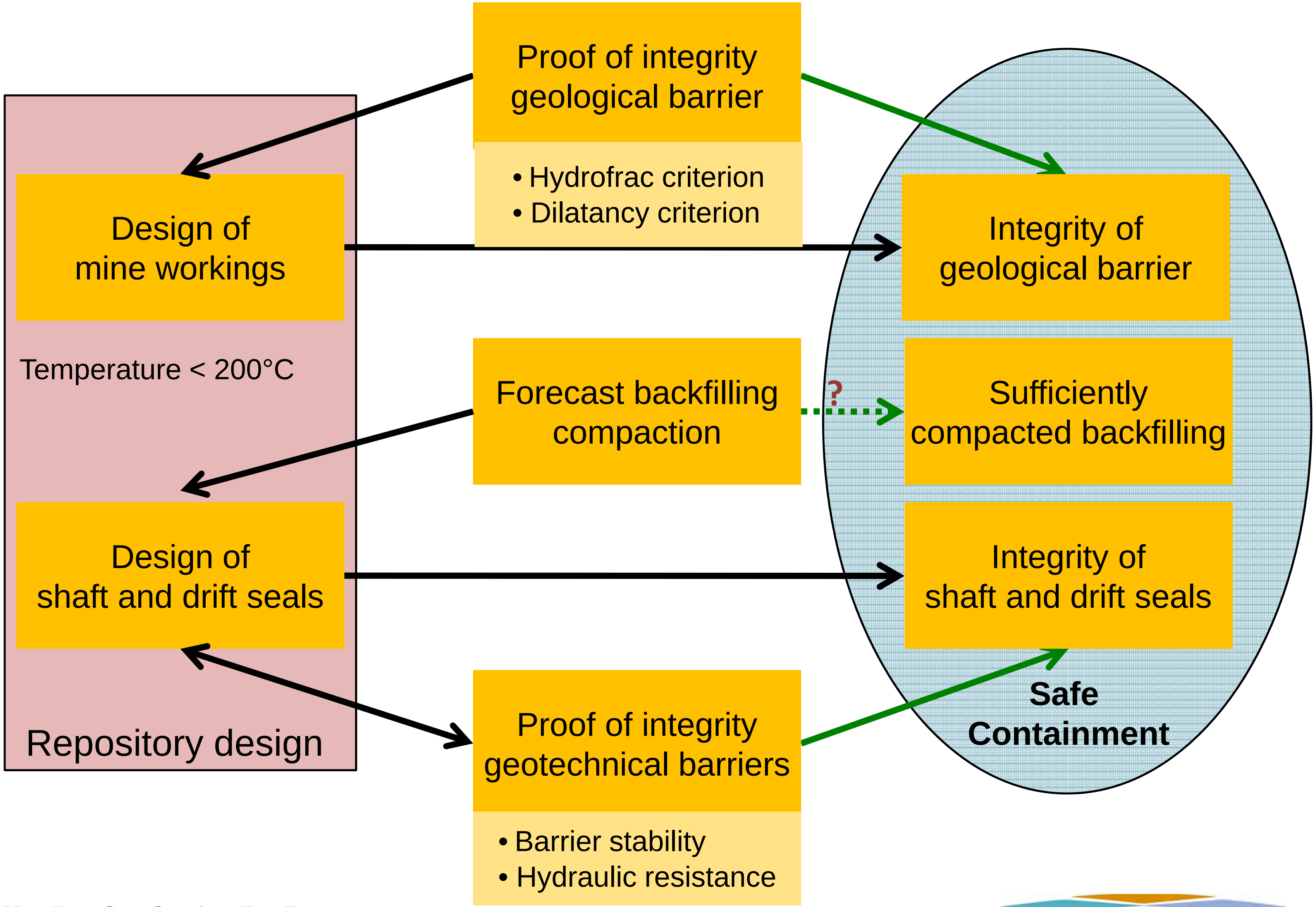
- Safe containment describes the condition of the repositoy system in which there is at the most an insignificant release of radionuclides from the containment providing rock zone during the demonstration period
- The containment-providing rock zone (CRZ) is that part of the repository system that ensures the containment of the waste
- An insignificant release from the CRZ is a release whose radiological consequences calculated by a biosphere model are below the permissible limits and thus pose no risk to subjections of protection



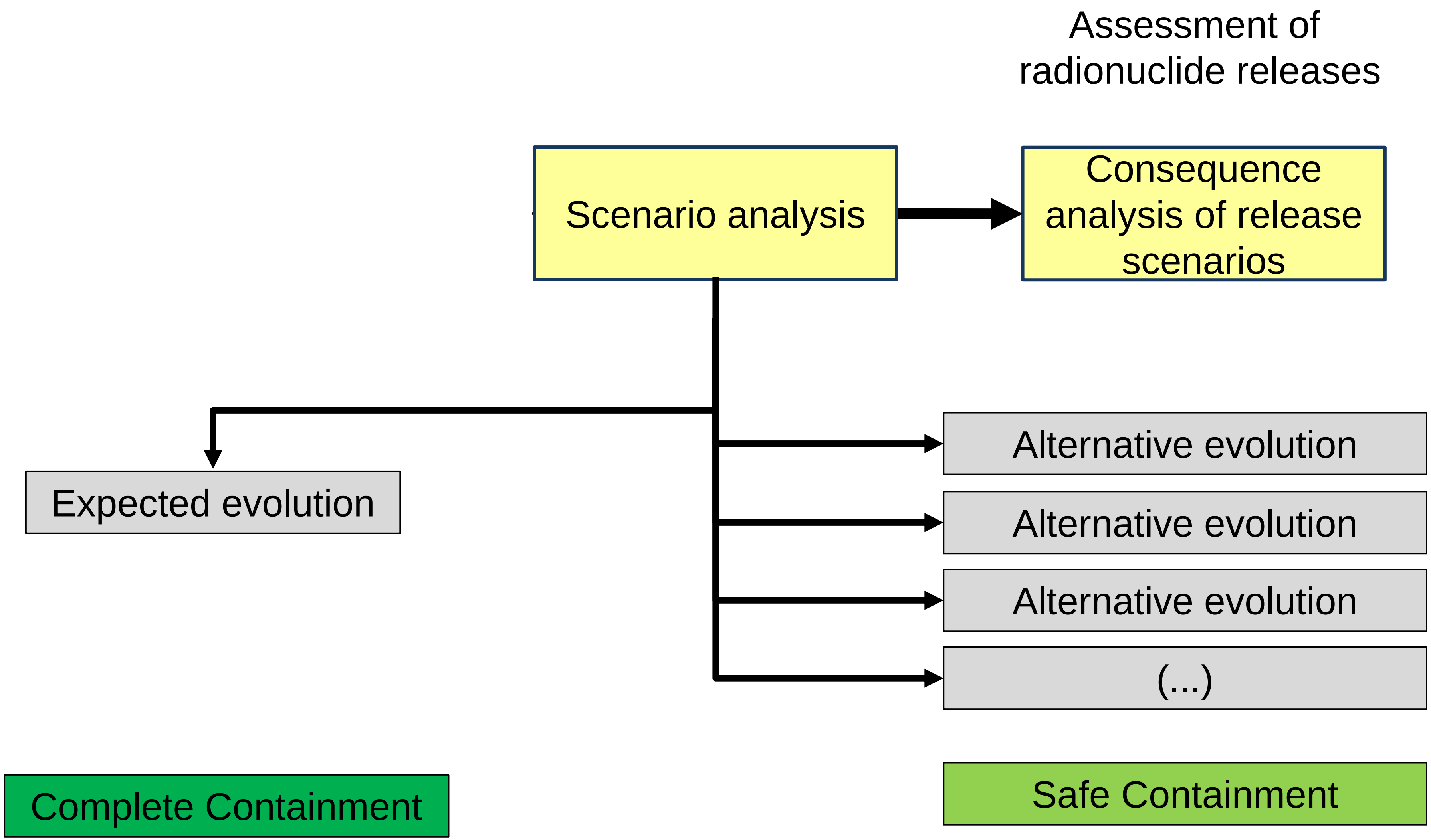
Functional requirements to demonstrate safety: How to reach safe containment?



Concept to demonstrate safety



Approach to demonstrate safety



Scenario Analysis

Development of a comprehensive FEP-catalogue

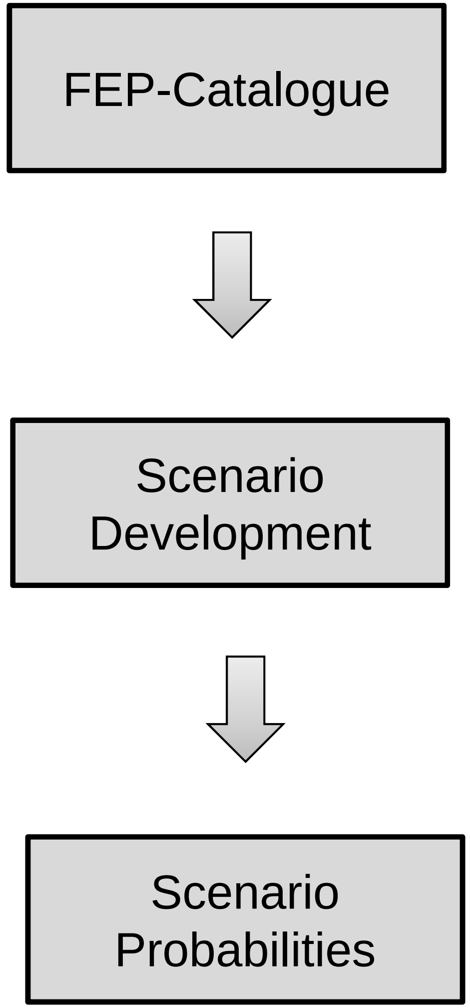
- Systematic description of each FEP
  - detrimental effects of containment relevant barriers
  - conditional probability of occurence
  - time frame of action
- FEP screening

Development of a set of scenarios

- Reference scenario
- Development of alternative scenarios

Derivation of scenario probabilities

- probable
- less probable
- can be excluded

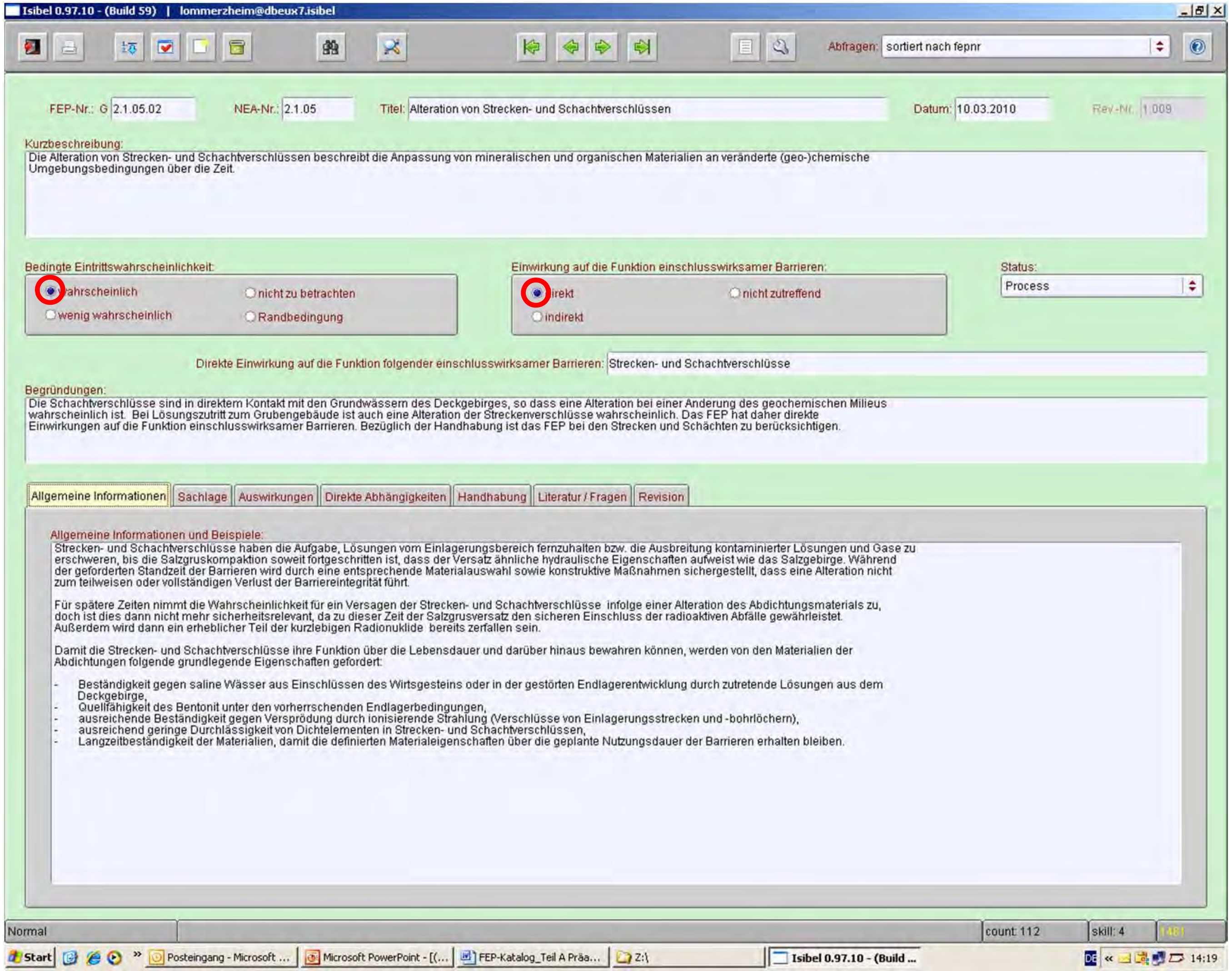


Development of a comprehensive FEP catalogue

- bottom-up and top-down approach to identify relevant FEPs
  - FEPs that have an impact on geological evolution
  - Possible evolutions that might cause in a release of radionuclides (→ FEPs)
- systematic description of each FEP
  - conditional probability of occurence
  - detrimental effects of containment relevant barriers
  - time frame of action
  - interaction with other FEPs

- realised in database format

Development of a FEP database



Scenario Development

- FEP-Catalogue provides the basics for the scenario development
- FEP-Catalogue is part of the approach to derive scenarios
- Reference scenario
  - probable FEPs
  - containment relevant barriers work as expected
- Alternative scenarios
  - probable and less probable FEPs
  - uncertainties
    - in the assumptions
    - in the characteristics of FEPs

Complete Containment

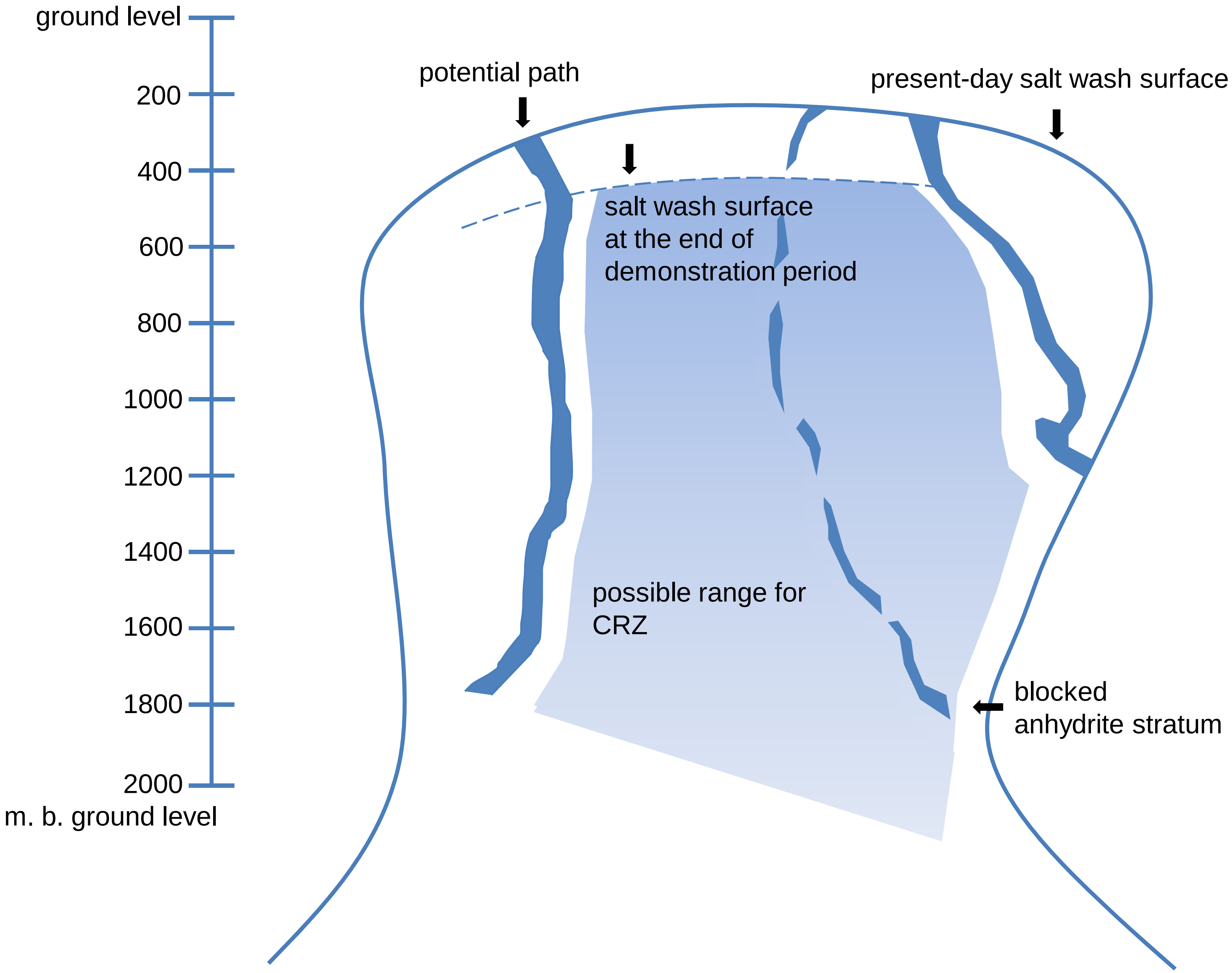
Safe Containment

How to proof if the containment is safe?

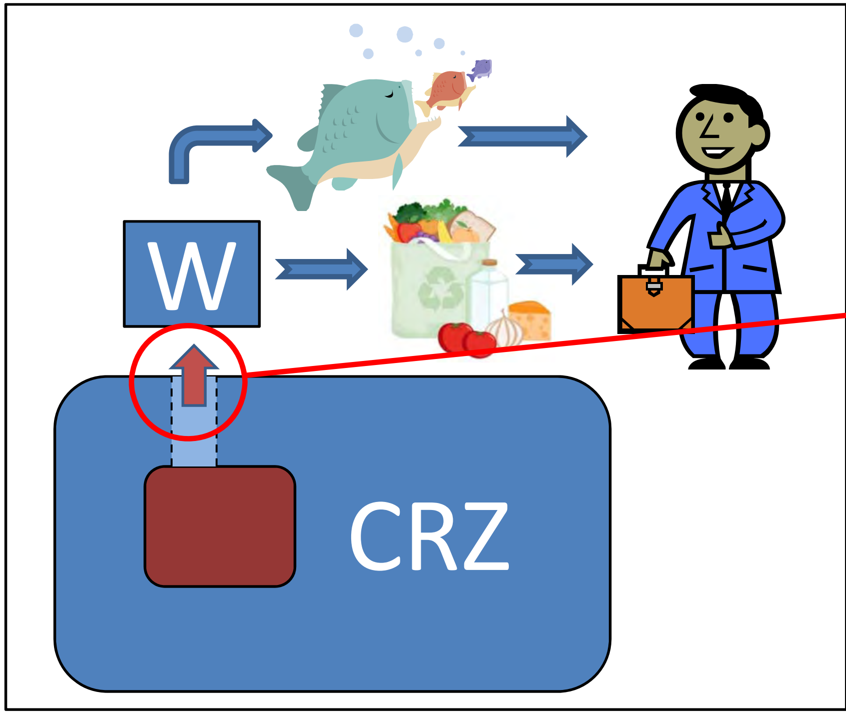
- Determination of a repository subsystem, that ensures containment → containment-providing rock zone (CRZ)
- Definition of indicators
  - at the boundary of the CRZ
  - relevant for human health → radiological significance
  - taking into account properties of salt formations → distinction between complete and safe containment Is there a pathway for radionuclides at all?

⇨ Proof of Safe Containment is a staged assessment with qualitative and quantitative arguments

CRZ in a salt dome



Radiological index of insignificance
Radiologischer Geringfügigkeitsindex (RGI)



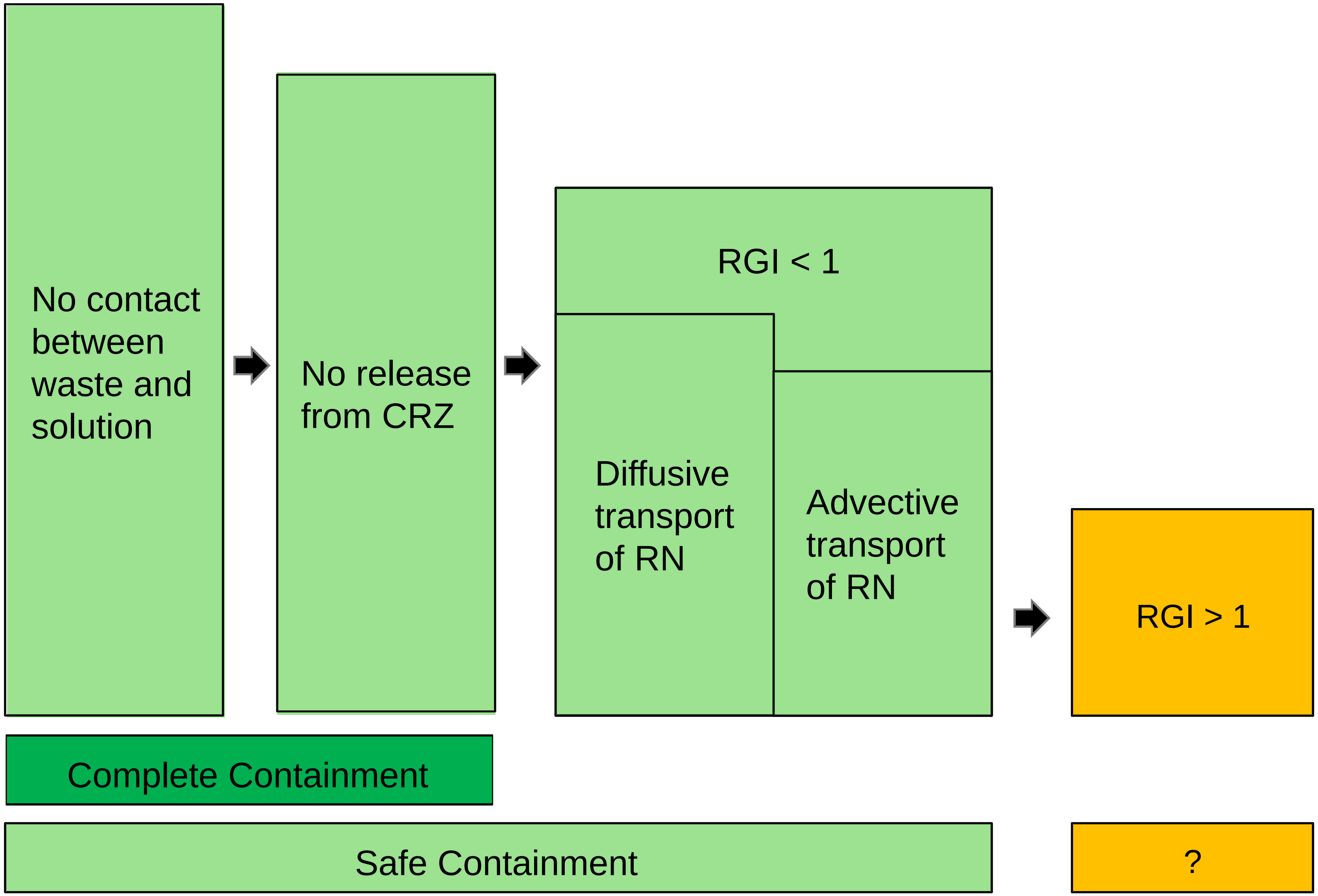
activity fluxes [Bq/a] — dose conversion factors

$$RGI = \frac{\sum_i S_i \cdot DCF_i}{W \cdot K_{RGI}}$$

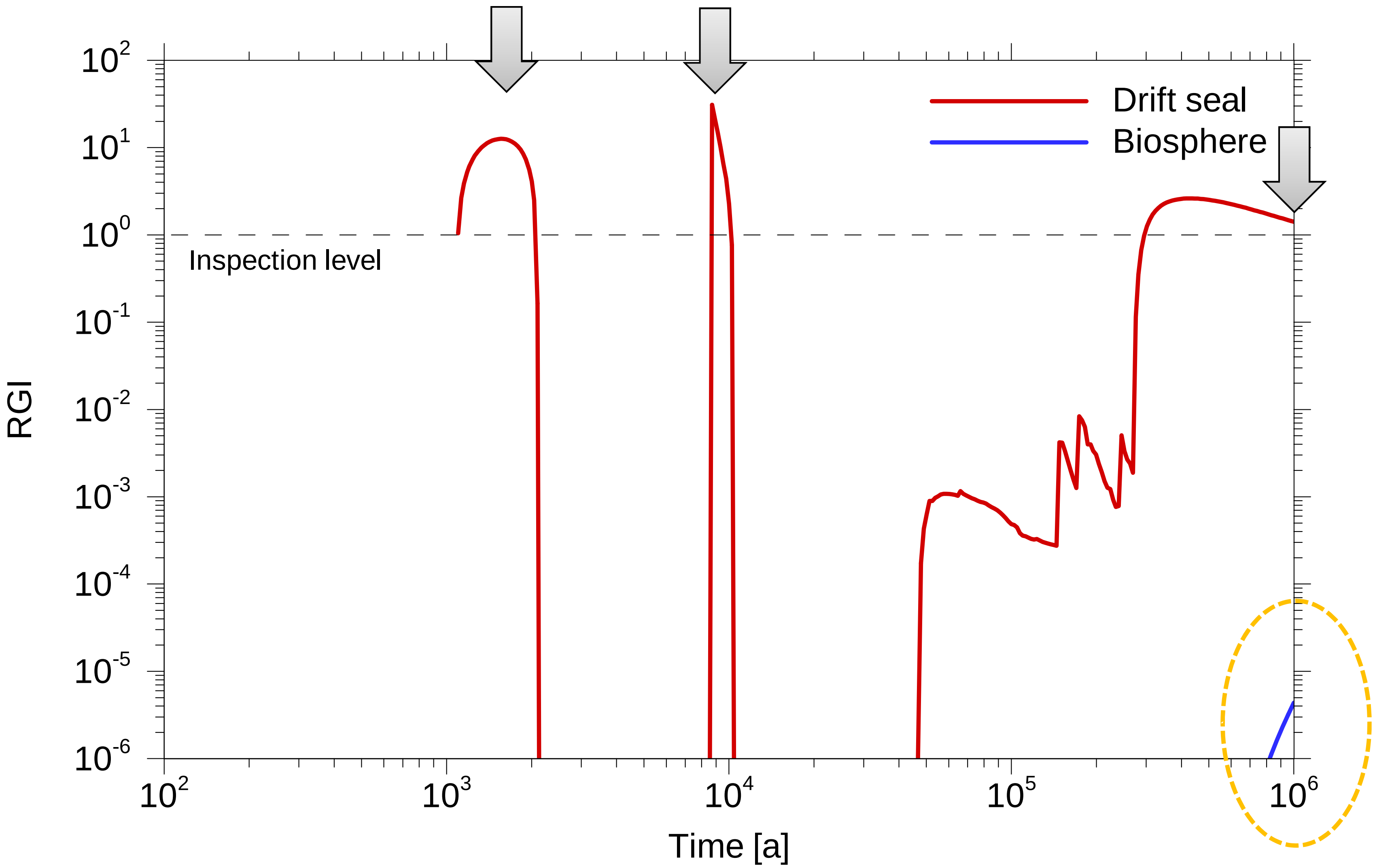
500 m³/a — inspection level: 0,1 mSv/a

- safety relevance, comparison with radiation exposure in biosphere
- stylised scenario is used

Method for the proof of safe containment



Example for a calculation of an "alternative evolution"



Conclusions

A novel concept to demonstrate safety has been developed, tested and shown to be applicable for the disposal of HLW in a repository in rock salt

- primary focus is proof the safe containment of the waste within CRZ
  - integrity of geological and geotechnical barriers
- proof of compliance with safety requirements for possible radionuclide releases

Methods have been developed and tested

- for developing scenarios (based on a comprehensive FEP catalogue)
- for proving integrity of geological and geotechnical barriers
- for addressing uncertainties comprehensively in the Safety Case
- for calculating radiological consequences of radionuclide releases from the CRZ
- for evaluating non-radiological consequences

Necessary R&D has been identified

Tools are at hand which are suitable to develop a Safety Case

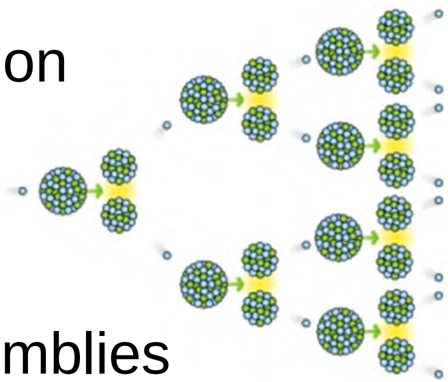
Robert Kilger (GRS)

Criticality Safety in the Waste Management of Spent Fuel from NPPs

Introduction

- **Criticality Safety Analysis** for **Spent Nuclear Fuel (SNF)**
  - Concept of "Burn-up Credit"
  - Requirements and Validation
- **Transport** and **Interim Storage**
  - Wet Pool Storage
  - Dry Cask Transport and Storage
- **Final Disposition**
  - Operational Phase
  - Post-closure Phase

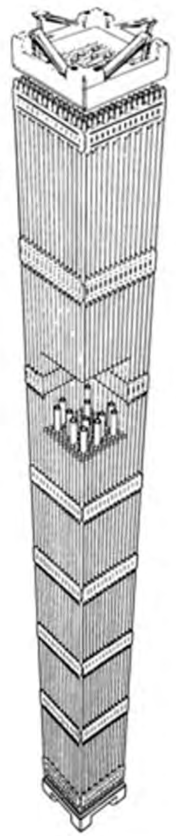
Criticality and Burn-up of Spent Nuclear Fuel



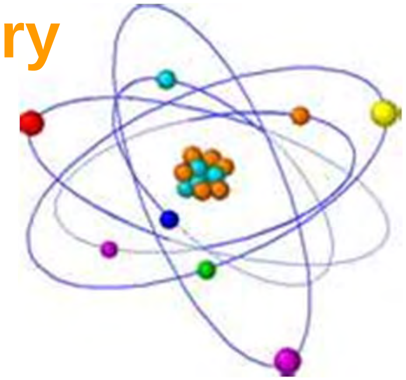
- **Criticality**: self-sustaining nuclear chain reaction
  - Neutron multiplication factor $k_{eff} = 1$
  - Mainly determined by nuclide inventory
- **Burn-up**: Integral measure for spent fuel assemblies
  - Energy produced per kg initial uranium [diff. fissions, (n,γ), …]
  - Determinable by measurement and by calculation
- **Nuclide inventory** of irradiated (spent) fuel
  - Loss of ^{235}U and ^{238}U
  - Production of fissile and absorbing Pu, absorbing fission products
  - Dependent on local neutron flux and height
  - Inventory not exactly determined by numerical value of burn-up

Criticality Safety Analysis: Burn-up Credit

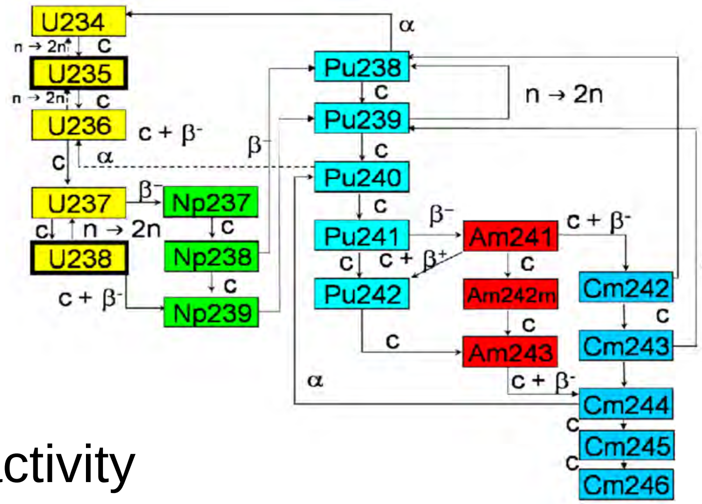
- Formerly: "**Fresh Fuel Assumption**"
  - No inventory determination required
  - Comparatively "simple" criticality calculation
  - Very large safety margin, often overconservative
- **Burn-up Credit**
  - Calculational approximation towards reality
  - Net consumption of fissile material
  - Generation of absorbing nuclides
  - Burn-up calculation for detailed inventory determination
  - PWR: Increase of enrichment, storage/transport capacity,…



Burn-up Credit: Nuclide Inventory



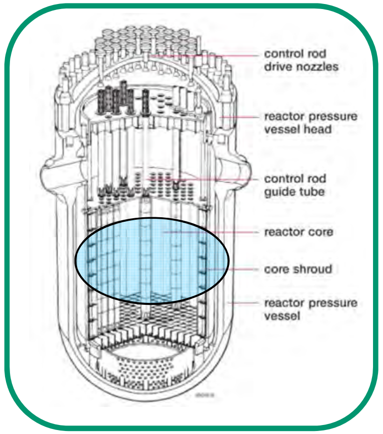
- **Fresh Fuel Assumption**: UO_2/MOX, water, structure materials
- **Burn-up Credit** (BUC)
  - Actinoides
  - Fission products, e.g. ^{149}Sm, ^{152}Sm, ^{143}Nd, ^{155}Gd, ^{133}Cs …

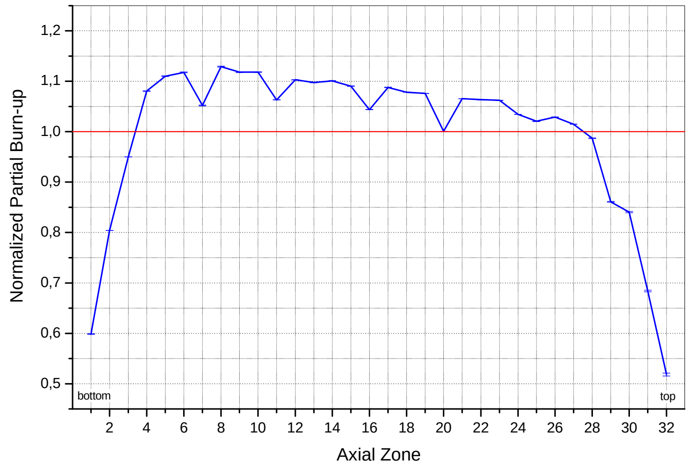


- **Requirements** for BUC nuclides
  - Fissiles: Consider all increasing reactivity
  - Absorbers: Only stable or long-lived nuclides, non-volatile also under accident conditions, may be considered
  - Validation

Axial Burn-up Profile

- **Axial Burn-up Profile** (ABP)
  - Due to neutron leakage, coolant temperature gradient, local and global neutron spectrum, …
  - Unique for each single spent fuel assembly





- Numerical k_{eff} with and without regard for ABP may differ
  - "**End effect**" $\Delta k = k_{axial} - k_{homogeneous} > 0$ for BU > ≈15 GWd/tHM

Validation of Calculation Methods

- **Burn-up Code** for inventory determination
  - Calculation method and cross sections
  - Calculation of post-irradiation radiochemical assay data
  - Detailed knowledge on (local) irradiation history required

- **Criticality Code** for k_{eff} determination
  - Calculation of critical benchmark experiments
  - Determination of systematic errors (bias) and error propagation
  - Sensitivity und uncertainty analysis

Wet Pool Storage

- Division of storage pool in two regions
  - Design of **region 1 for fresh fuel**

    Large dimensions

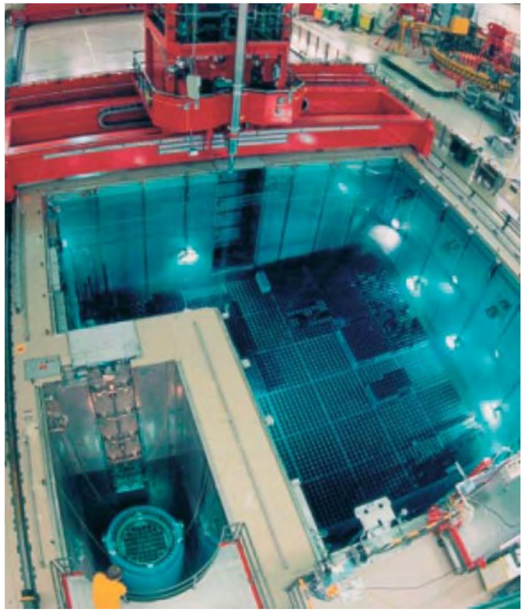
    Use of absorber plates
  - Design of **region 2 with burn-up credit**

    Dense assembly arrangement

    Control of assembly burn-up

    Protection against misload by redundant controls

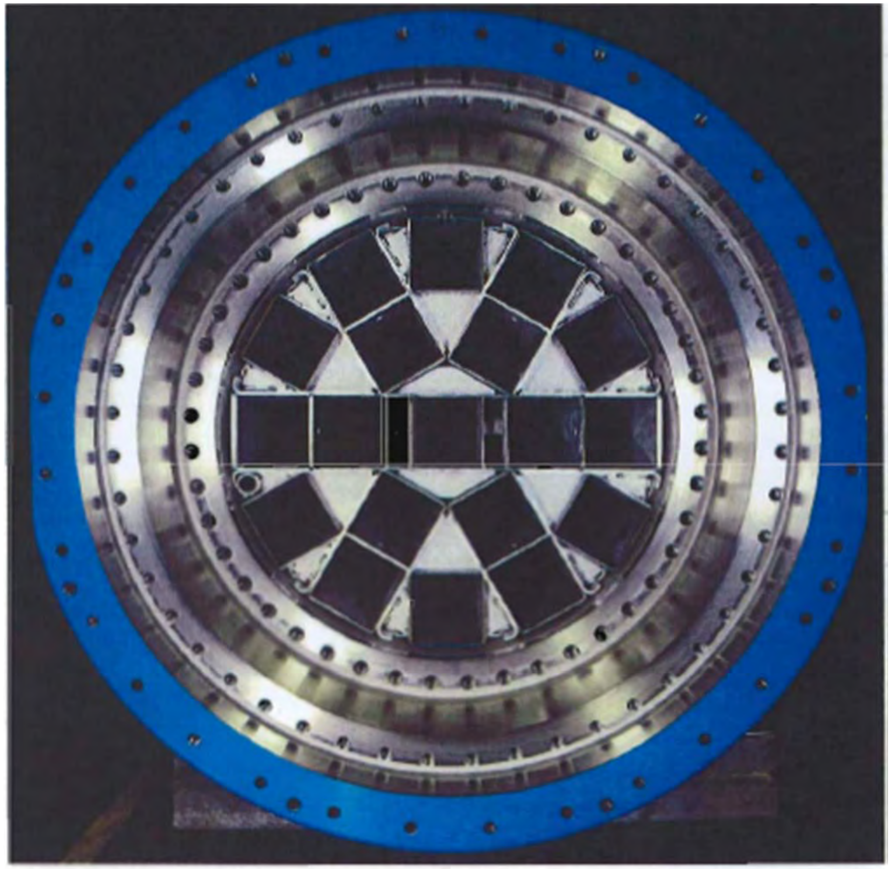
    Protection by design against single misloads



Source: NUKEM Technologies GmbH

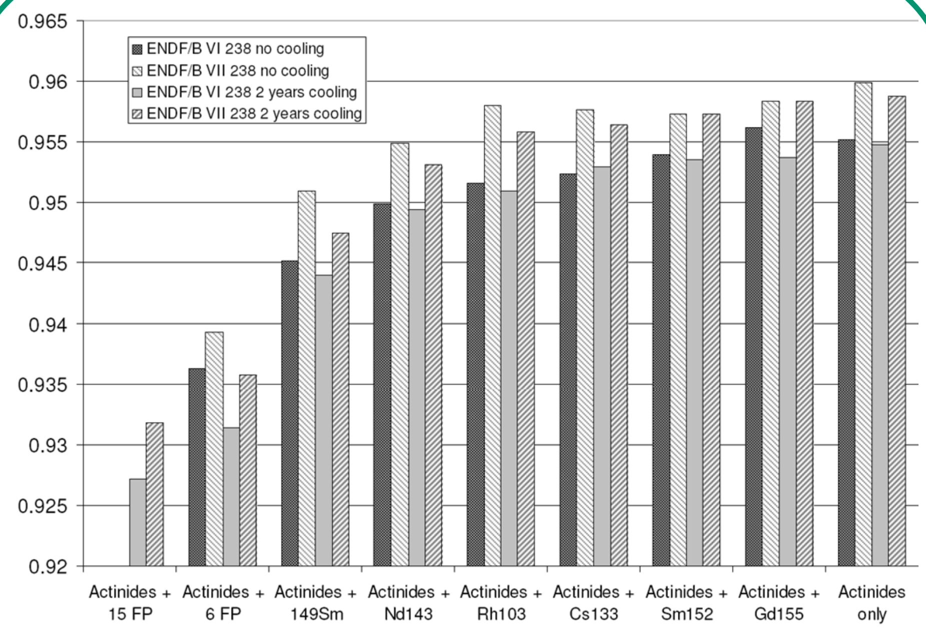
Transport and Storage Casks for Spent Nuclear Fuel

CASTOR V/19 from GNS



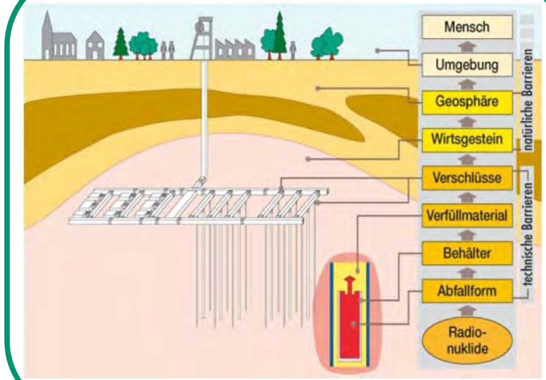
Transport and Storage Casks for Spent Nuclear Fuel

- Bundesamt für Strahlenschutz
  First license for a BUC transport and storage cask including six fission products in Germany expected in 2010
  - TN International TN24E
  - 12 GWd/tHM at 4.05 % ^{235}U initial enrichment



Source: Bundesamt für Strahlenschutz, NCSD2009

Criticality Safety for the Final Disposition



- **Operational phase** (cask emplacement)
  - Criticality safety given by design
- **Post-closure Phase**
  - Loss of control over repository after closure
  - Degradation of technical and natural barriers
  - Transition from deterministic to probabilistic analysis
  - Development of long-term evolution scenarios

    Water ingress (postulated in model scenarios)

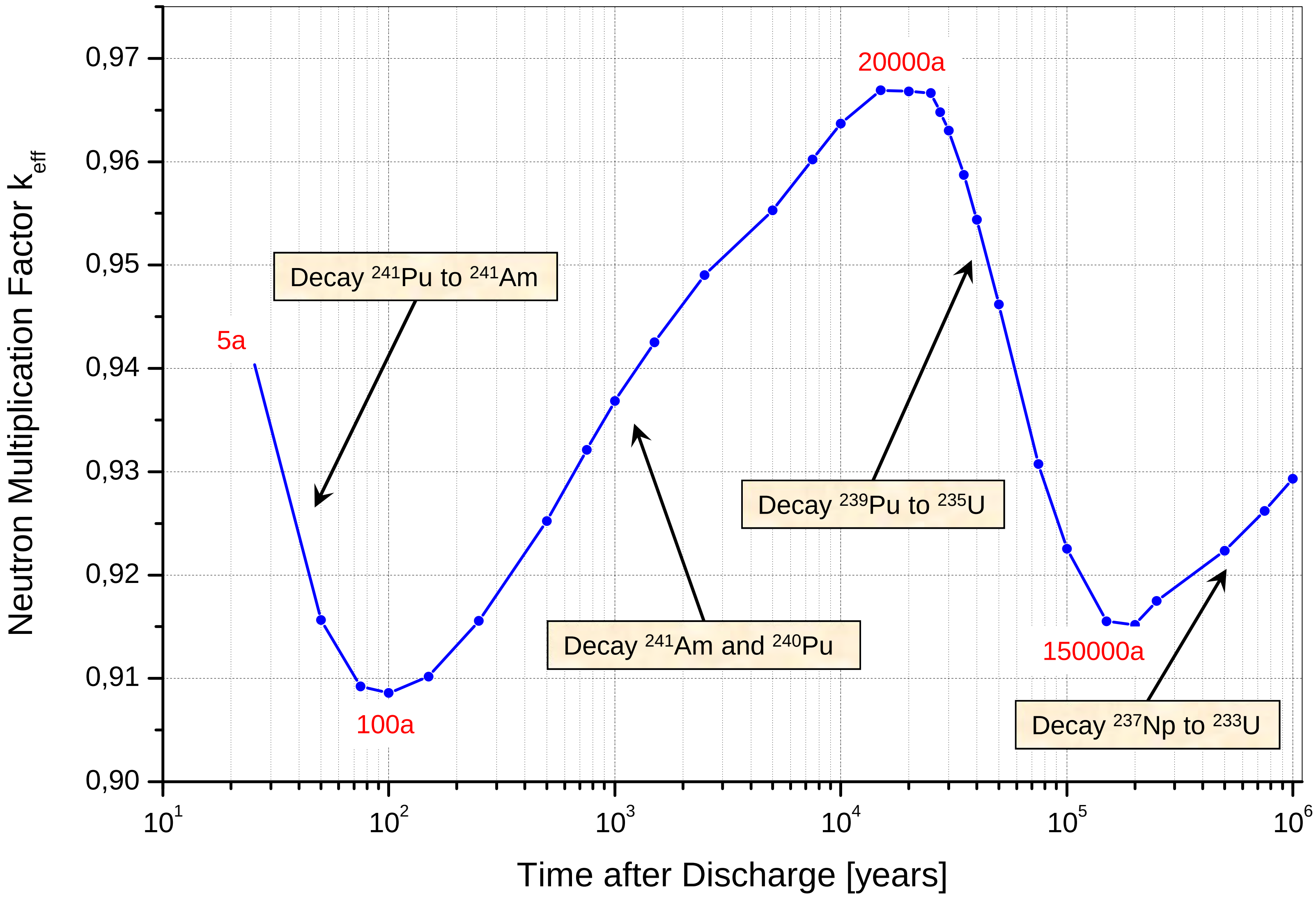
    Formation of uranium mineral phases

    Separation of uranium and plutonium

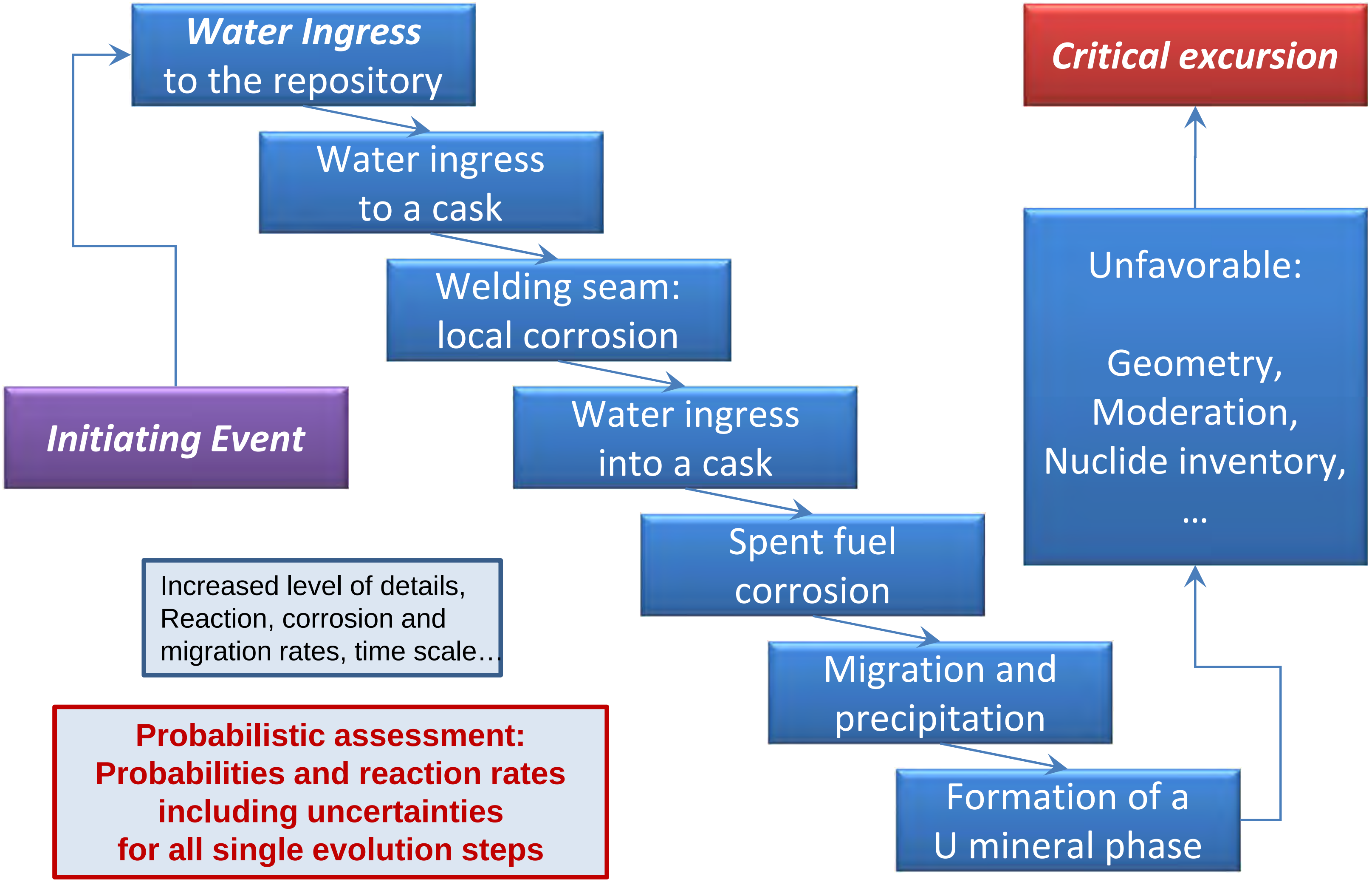
    Near field and far field

Pollux Disposal Cask
Source: GNS

Evolution of k_{eff} over Time for a Generic SNF System due to radioactive decay



Final Disposition: Exemplary Evolution Scenario



Generic Repository: Criticality Safety

- Possible repository evolution: **scenarios**
  - Water ingress crucial; in model postulated
  - Initiating event undefined in kind and point in time
  - Bandwidth of scenarios broadened by ABP
  - Supplementing probabilistic analyses
- Criticality based on inventory of a single cask extremely unlikely (this model: $P_{cond} < 10^{-6}$)
- Analysis of consequences
  - Excursion model for geological environments
  - Barrier tightness: Temperature, pressure, fission products…

Summary

- Remaining fissile material in spent nuclear fuel requires confirmation of criticality safety
- Assumption of fresh fuel very (overly) conservative
- Consideration of reactivity reduction by burn-up (“BUC”)
  - Strong requirements to code validation: Inventory, criticality
- Burn-up Credit applied in several countries, e.g. Germany
  - Wet storage pools, cooling ponds
  - Transport and interim storage in dry casks
  - Spent fuel dissolvers (reprocessing)
  - Inventory determination for geological long term scenarios

Thank you for your attention!

Erik Strub – Claudia Schmidt – Jörg Kaulard

Occupational Radiation Exposure of Workers in German NPPs

Content

- Introduction
  - Evaluation of Radiation Exposure Data
  - Dose Definitions
  - Situation in Germany
- Occupational Radiation Exposure in German NPPs
  - NPPs in Operation
  - NPPs under Decomissioning
  - Transition from Operation to Decommissioning
- Conclusion

Introduction - Evaluation of Radiation Exposure Data

- the evaluation of radiation exposure data important for experience feedback in the area of radiation protection
  - GRS operates a database (RADAN)
    - data on occupational exposure in German NPPs
    - data from operation and decommissioning
    - sources: annual reports, specific communications by the NPPs to international databases, like the database of the Information System on Occupational Exposure (ISOE)
- aim: long term trends of the occupational exposure in German NPPs, both in operation and under decommissioning

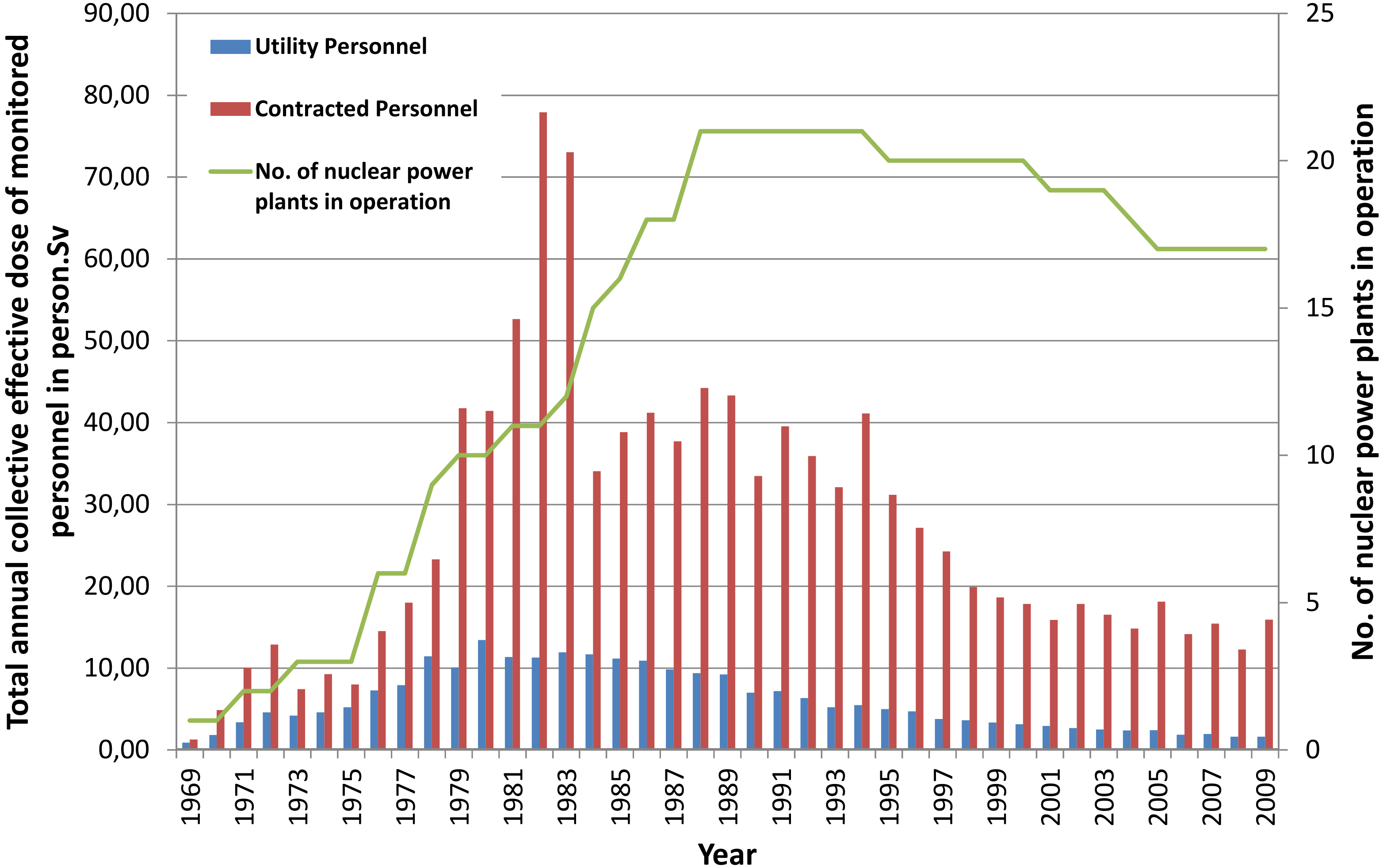
Introduction – Dose Definitions

- key data in this contribution
    1. annual individual effective dose
       (individual effective dose for 1 person in 1 year)
    2. average annual individual effective dose
       (average of all individual effective doses in 1 year)
    3. annual collective effective dose
       (sum of individual effective doses for 1 NPP in 1 year)
    4. total annual collective effective dose
       (sum of 1. for several NPPs)
    5. average annual collective effective dose
       (2., divided by number of NPPs)

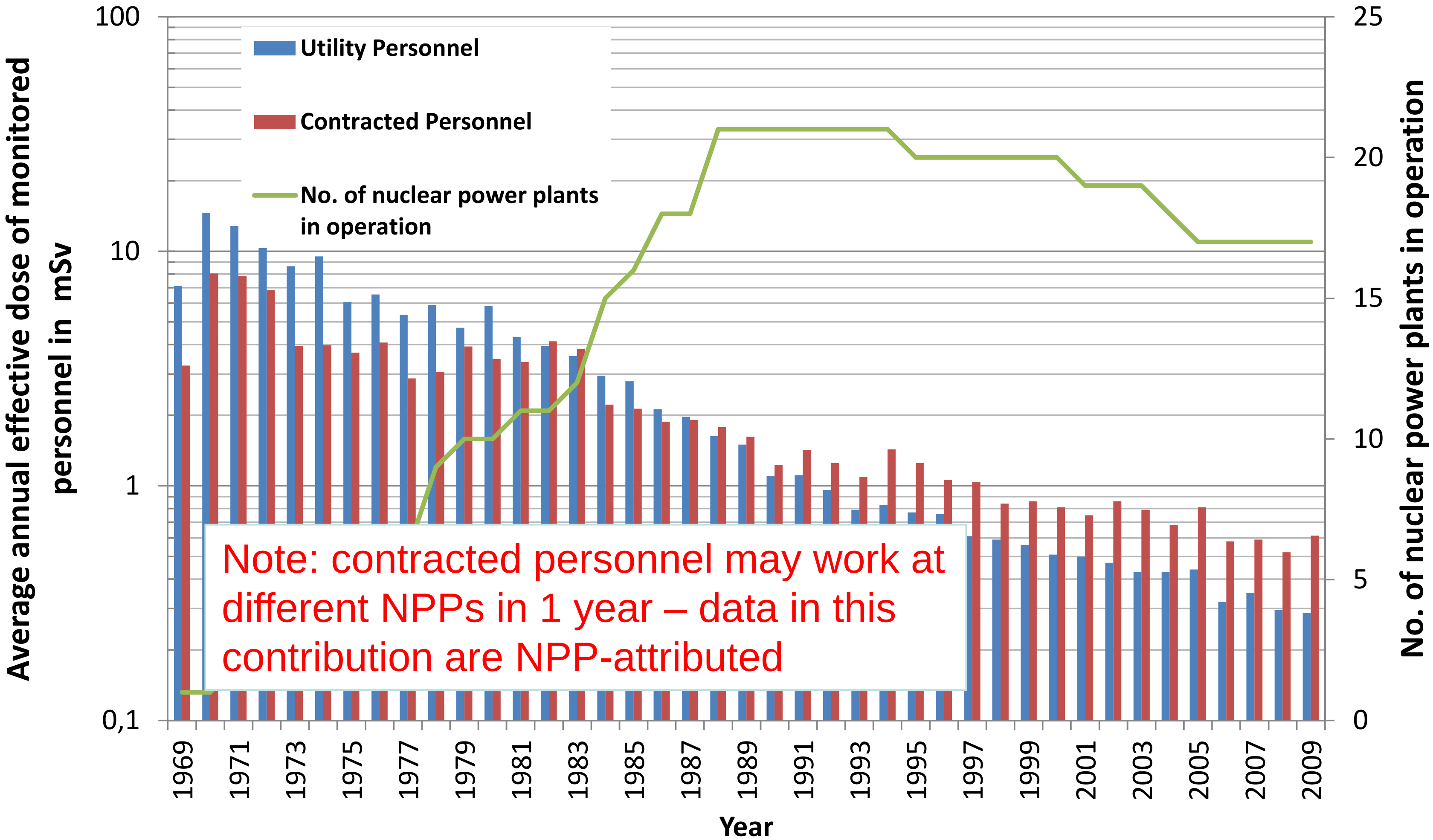
Introduction – the Situation in Germany

- in operation
  - 17 nuclear power plants at 12 sites
- under decommissioning
  - 17 nuclear power plants at 13 sites
  - 2 nuclear power plants completely dismantled

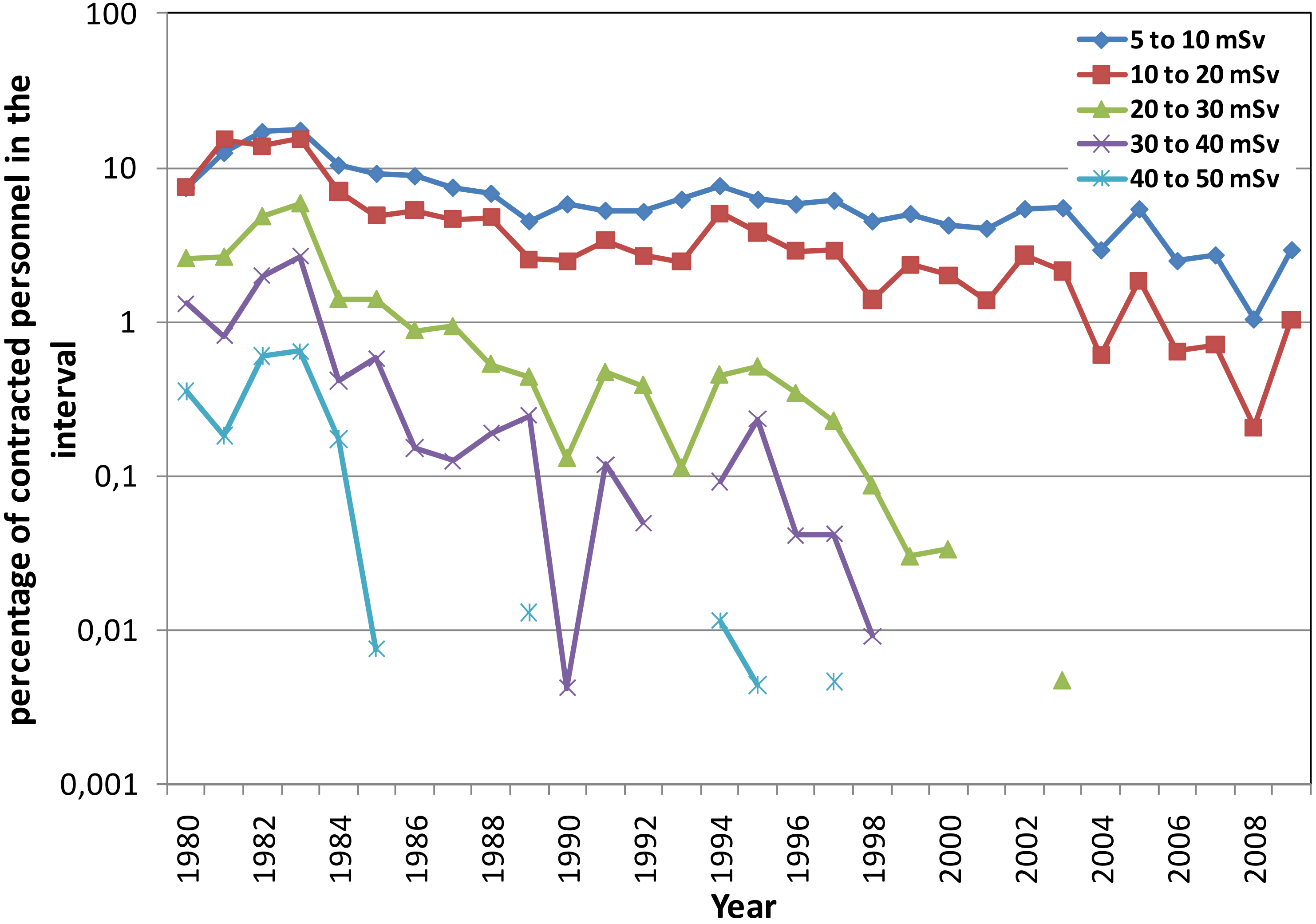
Occupational Radiation Exposure – German NPPs in Operation



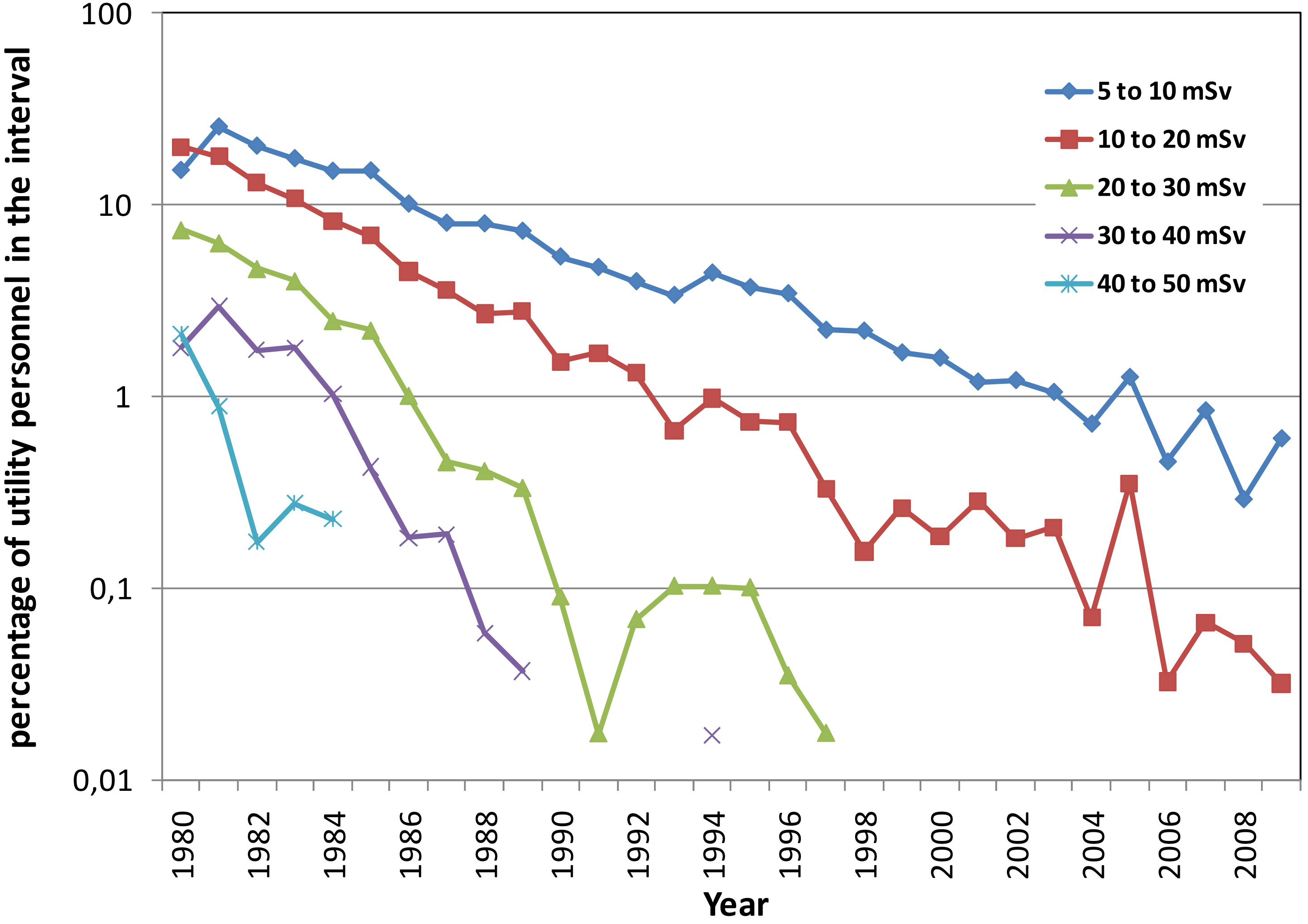
Occupational Radiation Exposure – German NPPs in Operation



Occupational Radiation Exposure – German NPPs in Operation



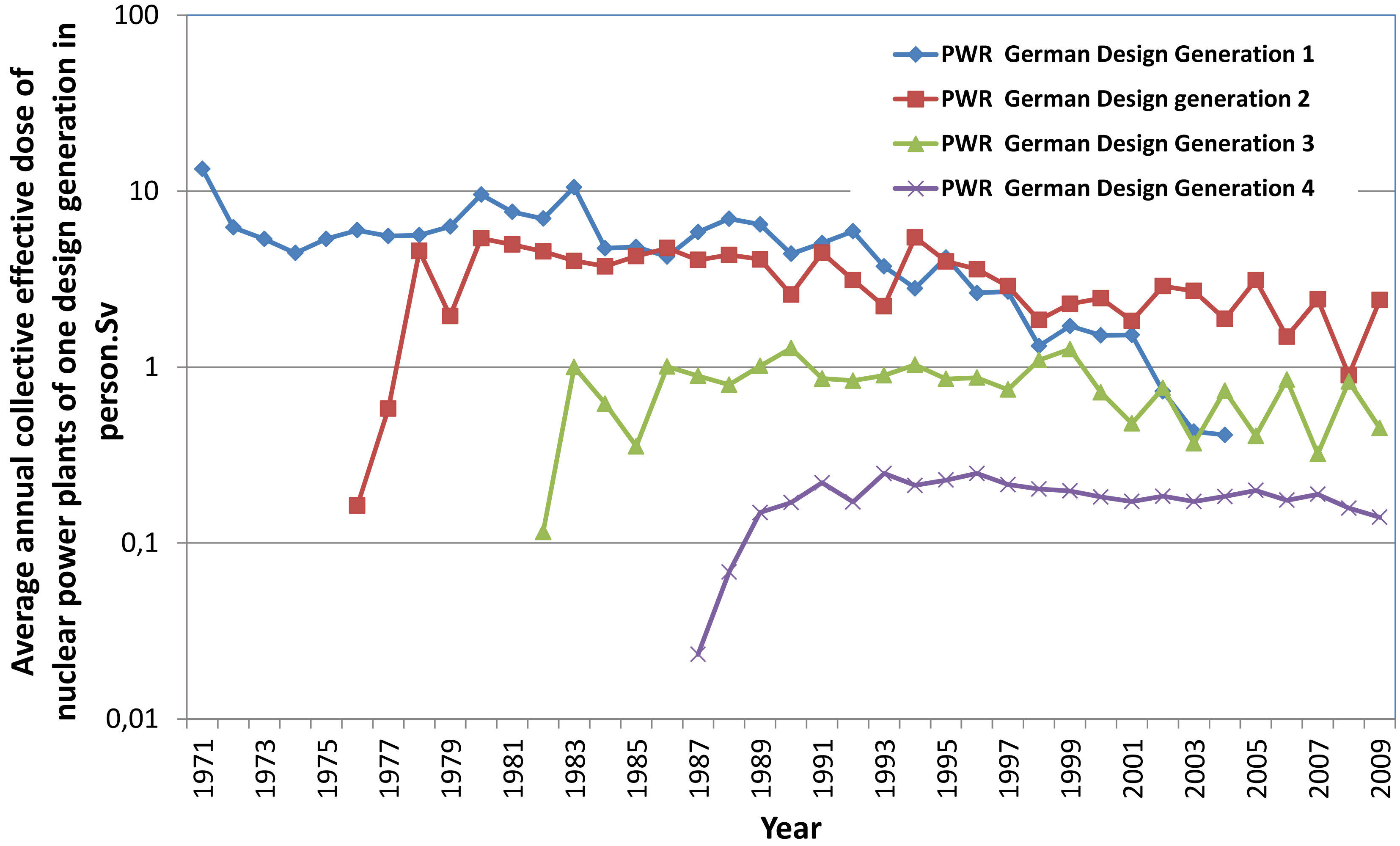
Occupational Radiation Exposure – German NPPs in Operation



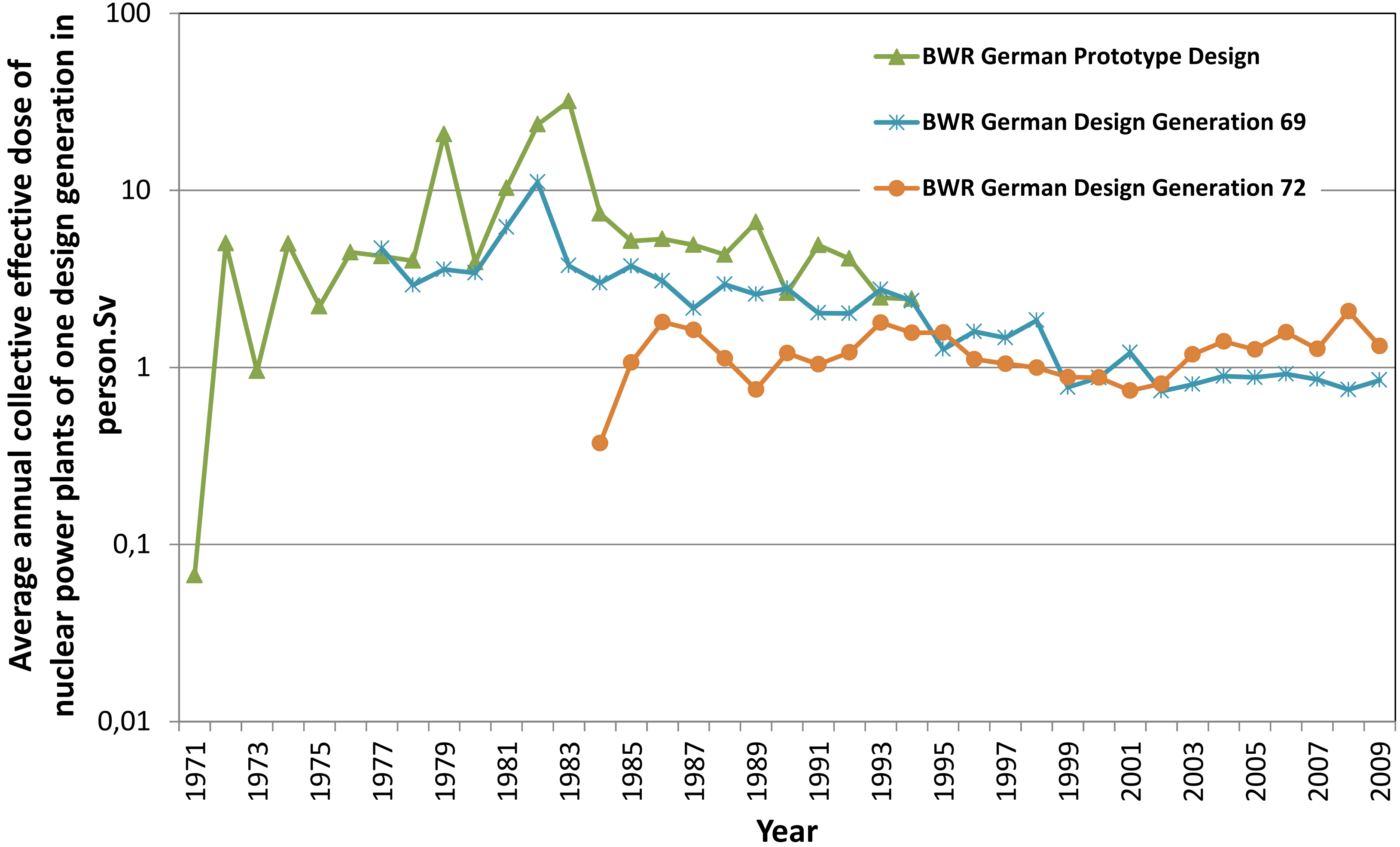
Occupational Radiation Exposure – German NPPs in Operation

- clear long term trend of lower *total annual collective effective doses* and *average annual individual effective doses* due to
  - consequent experience feedback
  - radiation protection work planning process during operation (based on the so called IWRS radiation protection planning)
  - improvements in the design of newer NPPs
  - some back fitting activities of older NPPs
- distribution of the *annual individual effective dose*
  - fraction of personnel with a higher dose decreases for both utility and contracted personnel
  - since 2001, exposures above the German dose limit (20 mSv per year) did not occur.

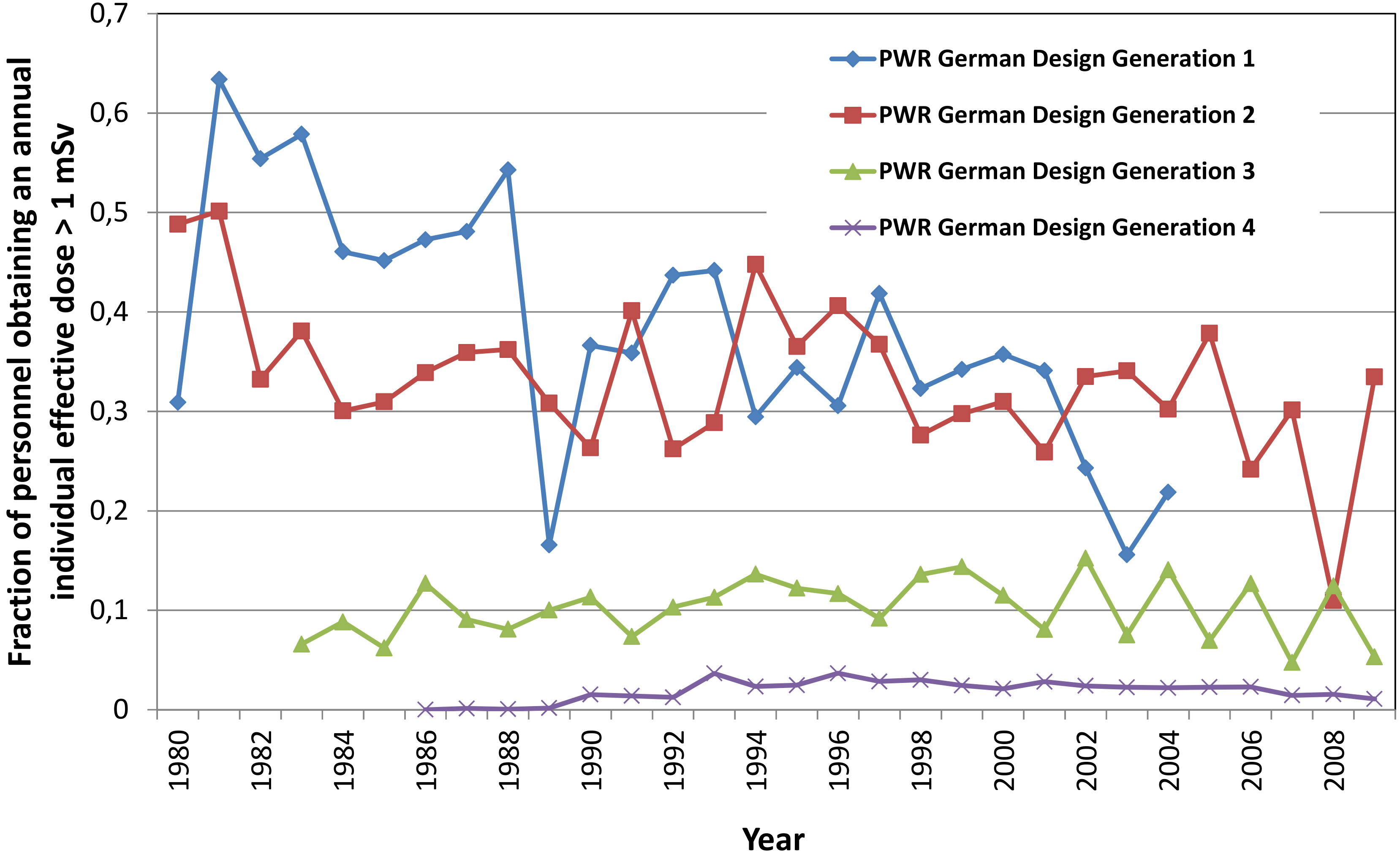
Occupational Radiation Exposure – German NPPs in Operation



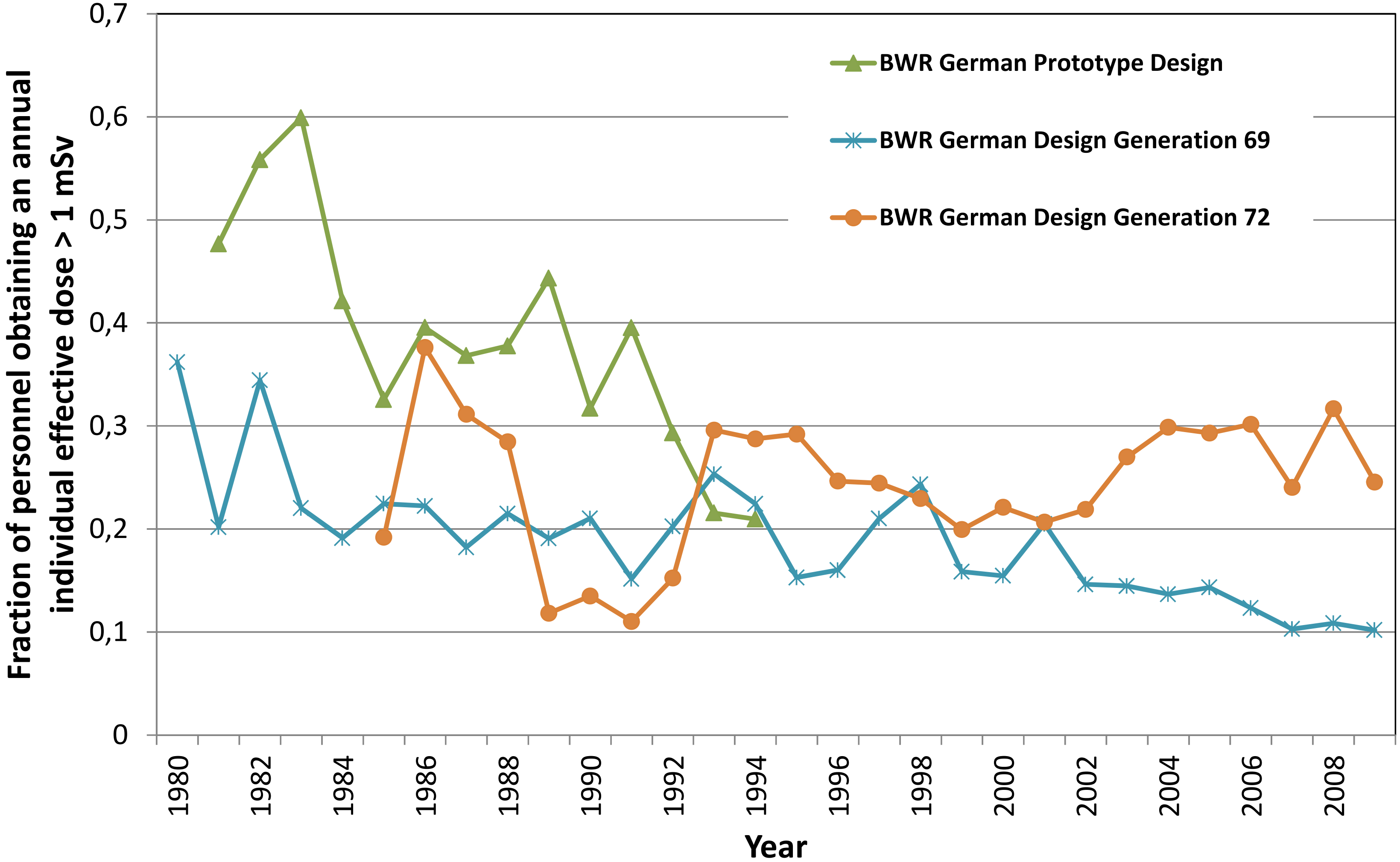
Occupational Radiation Exposure – German NPPs in Operation



Occupational Radiation Exposure – German NPPs in Operation



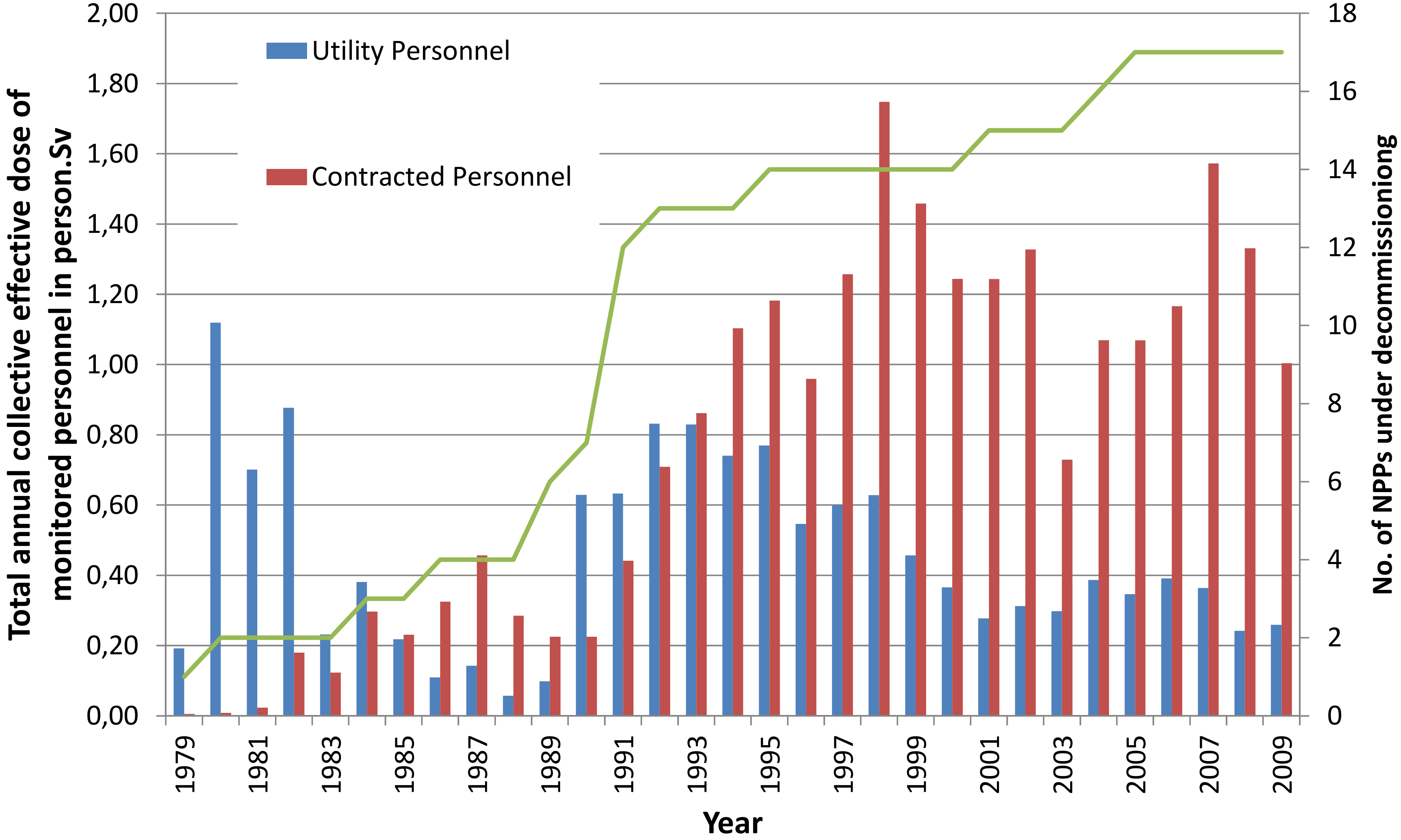
Occupational Radiation Exposure – German NPPs in Operation



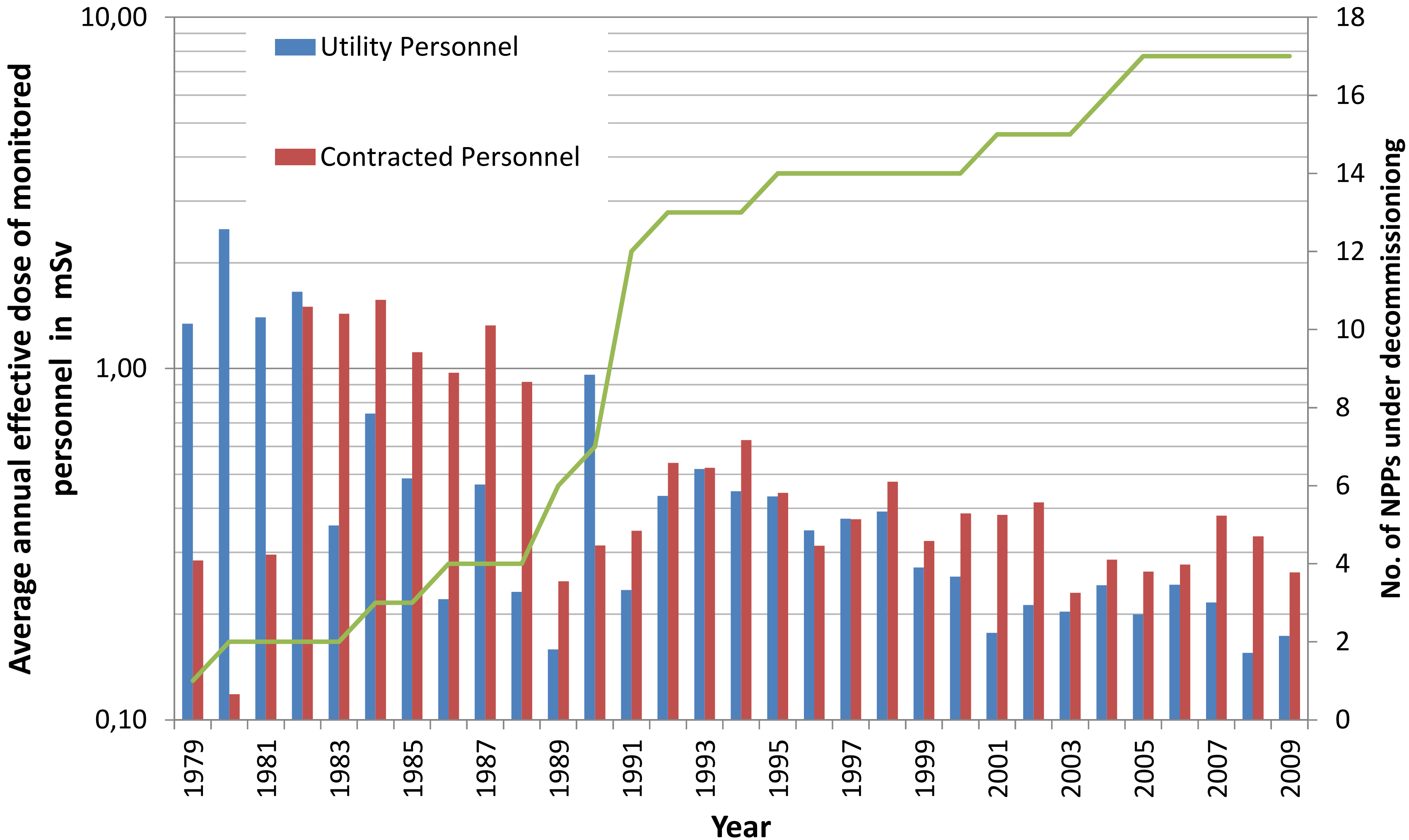
Occupational Radiation Exposure – German NPPs in Operation

- influence of the design of the German NPPs
  - during design of the PWRs of the fourth generation (Konvoi reactors) earlier experiences were considered
    - the use of material with low neutron activation
    - design of compartments to separate components with high dose rates from those with low dose rates
    - design requirements for low dose rates at frequently accessed locations fraction of personnel with a higher dose decreases for both utility and contracted personnel
  - generally, for all German design generations decreasing trends in the past

Occupational Radiation Exposure – German NPPs under Decommisioning



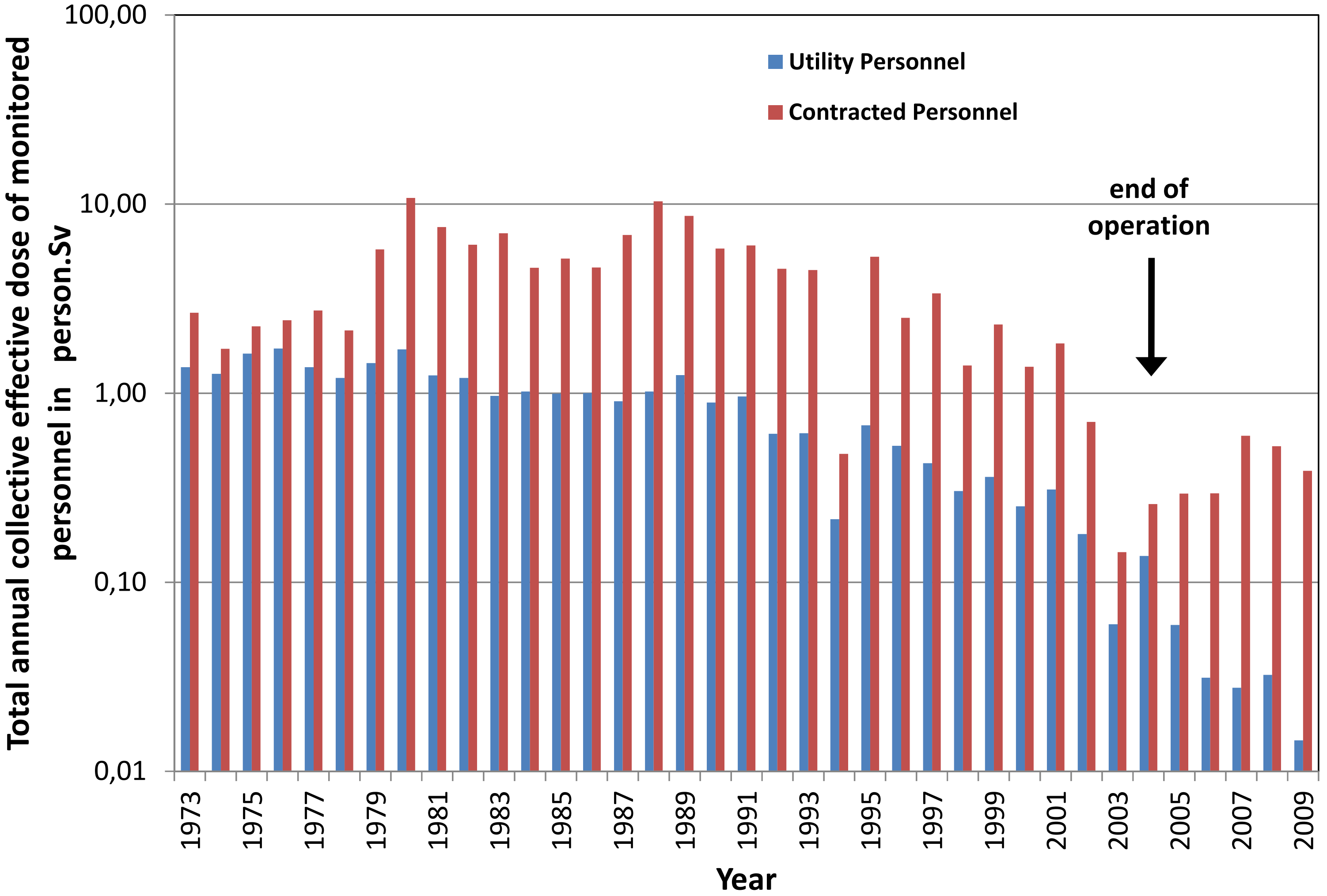
Occupational Radiation Exposure – German NPPs under Decommisioning



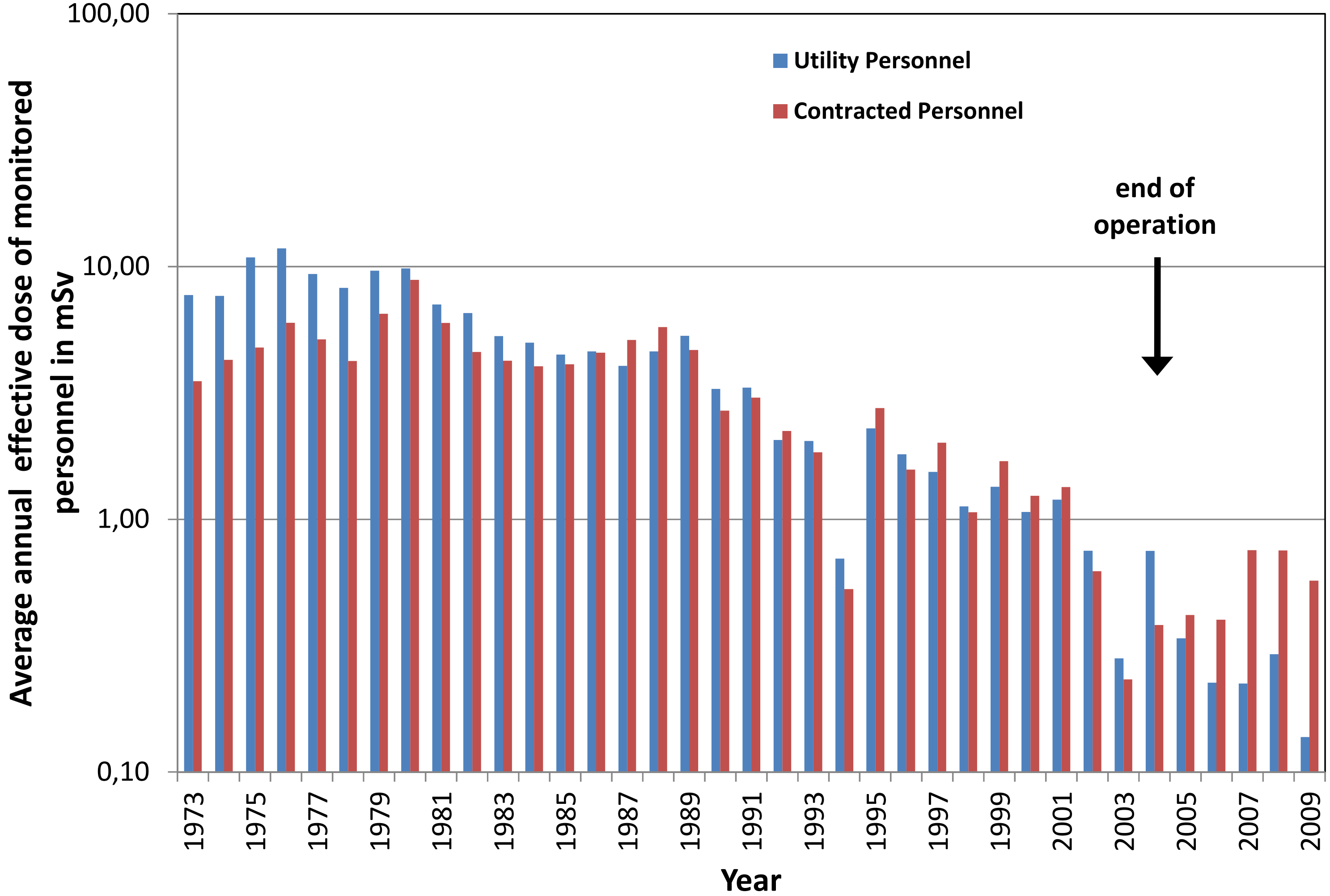
Occupational Radiation Exposure – German NPPs under Decommisioning

- no simple trends can be recognized
  - *annual collective effective dose* strongly depends on the decommissioning work and the related radiological conditions
  - work activities change from year to year, following the overall work planning and decommissioning strategy for the NPP
  - the type, inventory and operational history of the NPP influence the radiological conditions.
  - improvements e.g. due to experience feedback take place, but they can only be identified on the level of an individual NPP and only if the radiological conditions and the performed works are analyzed in detail.

Occupational Radiation Exposure – German NPPs (transition operation-decommissioning)



Occupational Radiation Exposure – German NPPs (transition operation-decommissioning)



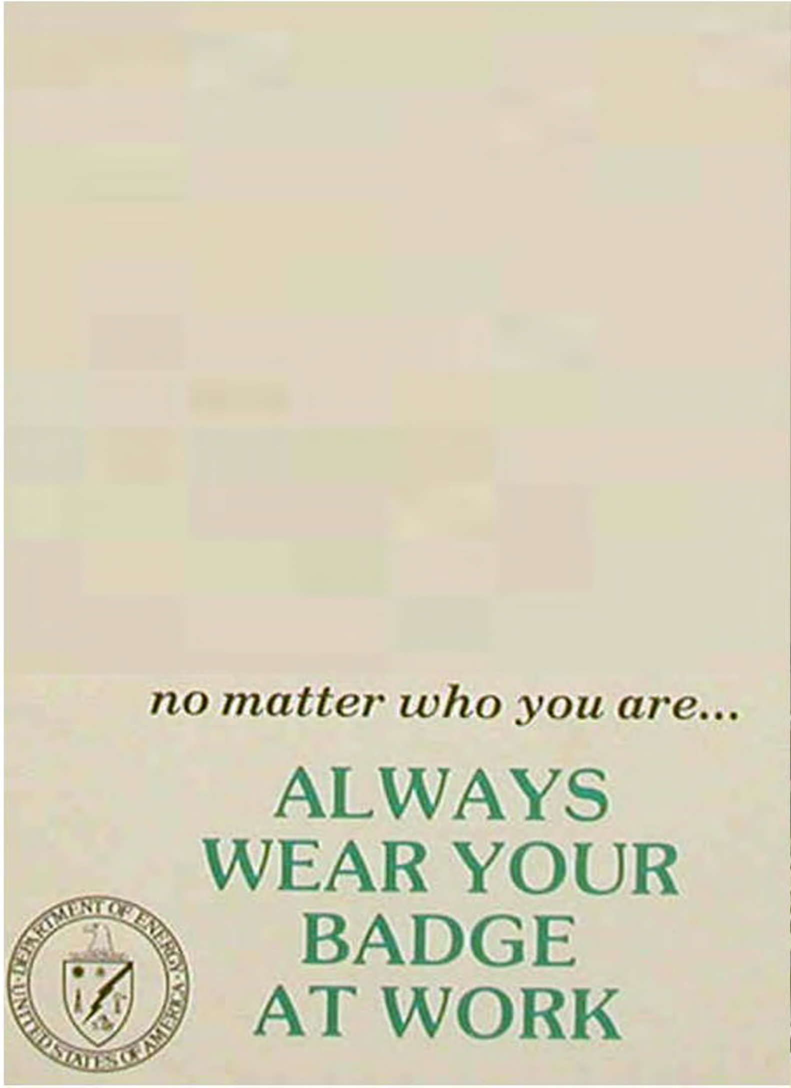
Occupational Radiation Exposure – German NPPs (transition operation-decommissioning)

- decommissioning related *average annual individual effective dose* is 10% to 20% with respect to operation (depending on the NPP and the work to be performed)
- dose reductions begin some years before the end of operation due to e.g.
  - reduced workload during the last outages
  - less improvement activities will be performed.
- during decommissioning, dose entities are changing from year to year
  - depending on the performed work
  - variations can not be interpreted without detailed knowledge about the decommissioning plan

Conclusions

- data on the occupational exposure during operation show decreasing trends
- improvements during operation depend on type/design generation of the NPP
  - improvements depending on radiological conditions and work to be performed
  - extent of improvement depending strongly on the design
  - higher savings in NPPs of the first design generations
- decommissioning NPPs
  - annual work changes essentially - no easy conclusion
  - improvements take place, but are difficult to account for without detailed knowledge

Thank you for your attention



International Conference

"Twenty-five Years after Chernobyl Accident. Safety for the Future"

Kyiv, *April 20-22, 2011*

Vladimir Kholosha, Deputy Minister of Emergency Management, Ukraine

26 April 2011 marks the 25th anniversary of the Chornobyl accident. In many countries, nuclear technology is regarded as one of the most effective solutions to meet the growing demand for energy, reduce greenhouse gas emissions, mitigate climate change and counterbalance fluctuations in prices for fossil energy sources. At the same time you can not forget the consequences of the Chornobyl accident.

The main purpose of the International Conference "Twenty-five years of the Chornobyl disaster: Security for the Future" to be held in Kiev on 20-22 April 2011, is to use the lessons of Chornobyl to ensure the safety of nuclear power and other dangerous technologies, and protecting people and the environment from emergencies catastrophic events.

International Conference

"Twenty-five Years after Chernobyl Accident. Safety for the Future"

Kyiv, *April 20-22, 2011*

Key topics to be addressed at the conference:

- *nuclear and radiation risks - cooperation of governments and people;*
- *consequences of nuclear and radiation accidents to human health and the environment;*
- *Shelter, removal of the Chornobyl NPP, the strategy of radioactive waste and spent nuclear fuel;*
- *development of prevention and response to nuclear and other man-made accidents, development of emergency plans, public awareness and involvement in emergency planning, and post-accident radiation monitoring;*
- *radiological consequences of the Chornobyl disaster, farming in contaminated areas, social and economic development of areas exposed to the Chornobyl disaster: successful development models, overcoming stereotypes and improving the investment attractiveness of regions;*
- *New technologies and scientific advances for future security.*

The languages of the conference are: Ukrainian, Russian, English.

International Conference

“Twenty-five Years after Chernobyl Accident. Safety for the Future”

Kyiv, *April 20-22, 2011*

Opening by Head of the International Organising Committee of the Conference 20/04 15-00

The statements from Conference co-organisers:

Statement from Ukraine, President of Ukraine Viktor Yanukovich

Statement from Belarus, President of Belarus Alexander Lukashenko

Statement from Russia, President of Russia Dmitry Medvedev

Statement from UN, Mr. Ban Ki-moon, Secretary-General of the United Nations

Statement from Council of Europe, Mr. Thorbjørn Jagland, Secretary General of the Council of Europe

Statement from EC, Mr. José Manuel Barroso, President of the European Commission

Statement from IAEA Mr.Denis Flory, Deputy General Director, IAEA

Keynote speeches by high level representatives of the Governments and International organisations:

Mr. Norbert Röttgen, Federal Minister for the Environment, Nature Conservation and Nuclear Safety, Germany

Mr. Jean-Louis Barloo, Minister for Ecology, Energy, Sustainable Development and Town and Country Planning (Regional Development), France

Mr. Thomas Mirow, President of the European Bank for Reconstruction and Development

Mrs. Ann-Louise Eksborg, Director General of the Swedish Radiation Safety Authority

Mr. Jan Mans, President of the European Forum for Local and Regional Disaster Management

International Conference

“Twenty-five Years after Chernobyl Accident. Safety for the Future”

Kyiv, *April 20-22, 2011*

April 21, 2011 9:00-13:00 Invited report

1. “Radiological Risk: the Need to be Informed, the Right to be Protected”, Thorbjørn Jagland, Secretary General of the Council of Europe
2. “Chernobyl: Lessons of Safety”, Vladimir Kholosha, Deputy Minister, Ministry of Emergency Management, Ukraine
3. “International Support Towards the Mitigation of the Chernobyl Accident Consequences”, Frank-Peter Weiss, Technical and Scientific Director of GRS, Germany
4. “Nuclear Safety in the 21st Century : reflecting on the lessons learnt “ Mr. Jacques Repussard, Director General of IRSN, France
5. “Object “Shelter” – Prospective and Challenges” Igor Gramotkin, Director Chernobyl NPP, Ukraine
6. “ Radiological and Medical Consequences of Chernobyl accident. Lessons Learnt” Vladimir Bebeshko, Director SRCRM, Ukraine + WHO (tbd)
7. “Strengthening of Nuclear Safety and Radiation Protection, Lessons Learned from Chernobyl.” Olga Makarovskaya, Deputy Head of UNRC, Ukraine + IAEA (tbd)
8. “Chernobyl and New Knowledge” Vyacheslav Shestopalov, Ukraine, Viktor Poyarkov, (TESEC), Hartmuth Teske, (Germany), Michel Chouha, (France)

International Conference

“Twenty-five Years after Chernobyl Accident. Safety for the Future”

Kyiv, *April 20-22, 2011*

April 21, 2011

14-00 18-00 Sectional sessions on conference main subjects

April 22 2011,

9:00-12:00, Conference Conclusion, recommendations for the future.

- · ***Conclusions of sectional sessions***
- · ***General conclusions of the conference and recommendations for the future.***
- · ***Closing of a conference***
- · ***Press-conference***

THANK FOR YOUR ATTENTION! WELCOME TO CONFERENCE!
Complementary informations : vnd@icz.com.ua

IAEA Support for the Establishment of Nuclear Security Education

Andrea Braunegger-Guelich, Vladimir Rukhlo

Office of Nuclear Security

International Atomic Energy Agency

Department of Nuclear Safety and Security

EUROSAFE Conference,

8-9 November 2010, Cologne, Germany

IAEA

International Atomic Energy Agency

Contents

- Nuclear Security
- International Instruments
- Nuclear Security Series No. 12- Educational Programme in Nuclear Security
- International Nuclear Security Education Network
- Conclusion

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Nuclear Security

Prevention **Detection** **Response**

Why is Nuclear Security today an international concern?

- Theft of nuclear weapon
- Theft of material to make improvised nuclear explosive device
- Theft of radioactive material for radiological dispersal device
- Sabotage of facility or transport

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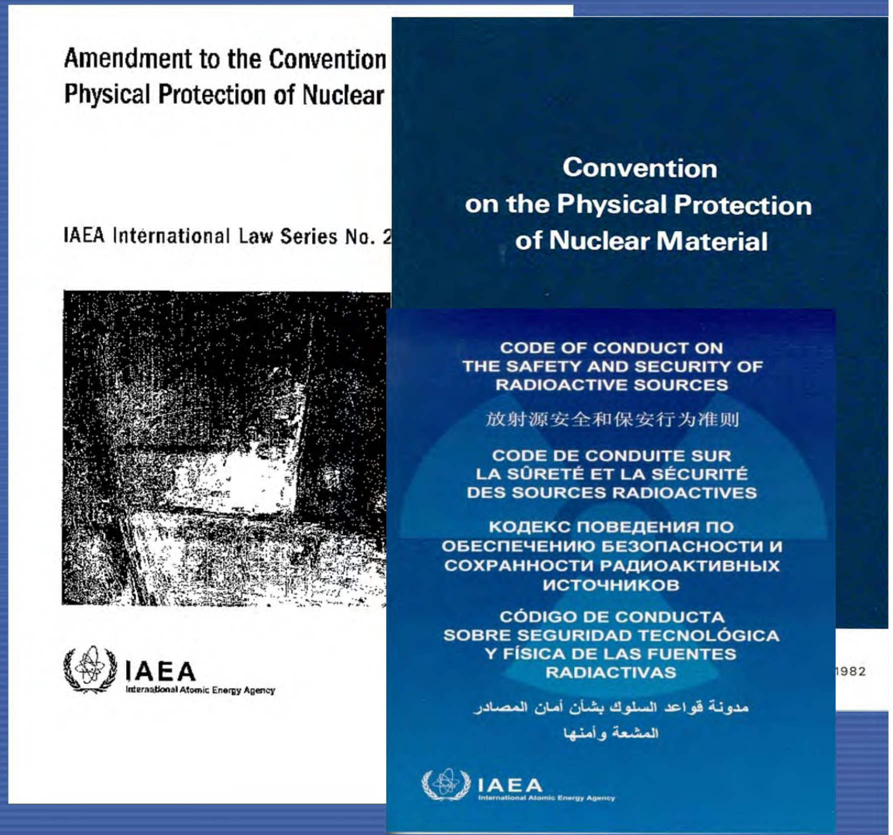
International Instruments

Legally binding:

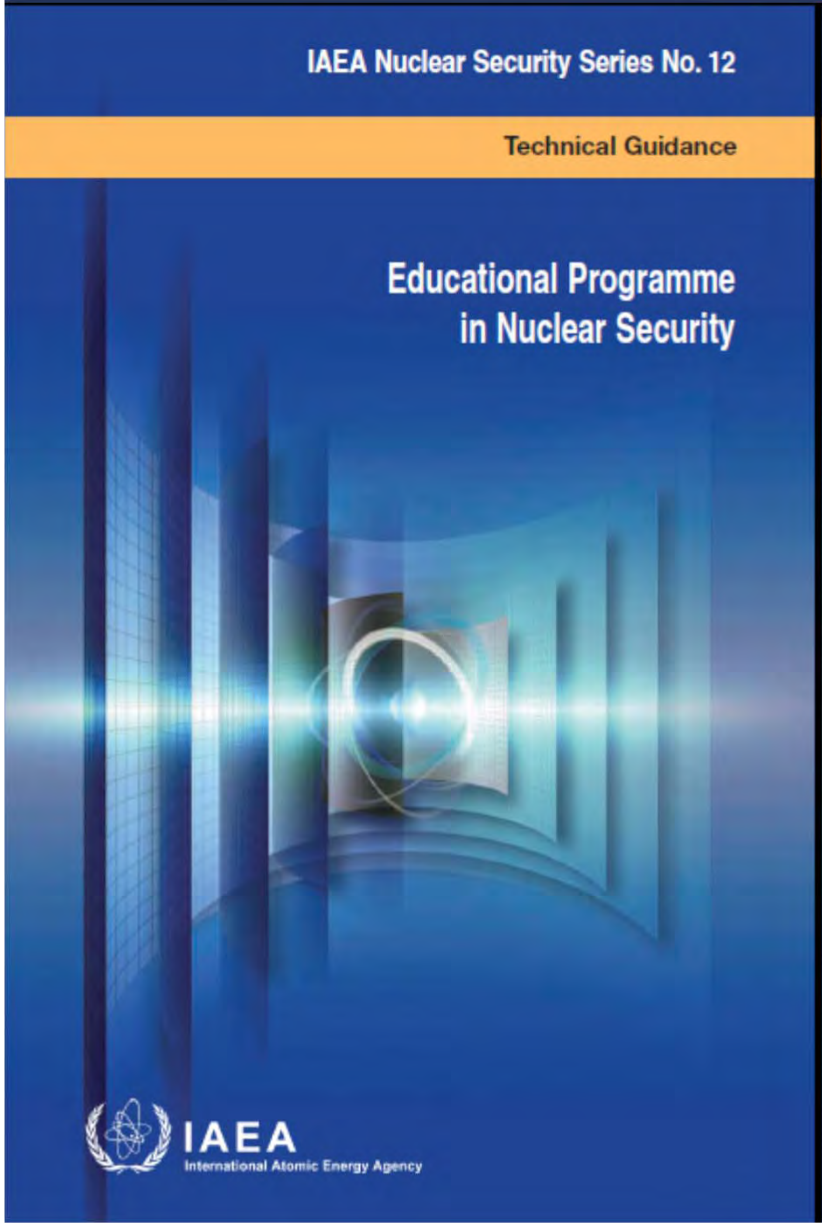
- Convention on the Physical Protection of Nuclear Material & Amendment
- Safeguards agreements and additional protocols
- Convention on the Suppression of Acts of Nuclear Terrorism
- Security Council resolution 1540
- Security Council resolution 1373

Non-binding:

- Code of Conduct on the Safety and Security of Radioactive Sources



Nuclear Security Series No. 12



The Nuclear Security Series No. 12 – Educational Programme in Nuclear Security was developed following the IAEA standard procedures.

It was published in April 2010.

It can be downloaded:

http://www-pub.iaea.org/MTCD/publications/PDF/Pub1439_web.pdf

Nuclear Security Series No. 12
Objectives

- Support nuclear security sustainability in States
- Prepare professionals to carry out nuclear security assignments necessary to meet any obligations under the global nuclear security framework
- Provide guidance for developing a Master of Science & a Certificate Programme in Nuclear Security
- Provide a comprehensive and current overview of nuclear security

Nuclear Security Series No. 12 Content

- **Four main sections**
  - Introduction
  - Capacity Building in Nuclear Security - Human resource development
  - M.Sc. Programme in Nuclear Security
  - Certificate Programme in Nuclear Security
- **Two appendices**
  - Recommended Courses for M.Sc. Programme in Nuclear Security
  - Appendix II. Recommended Courses for Certificate programme in Nuclear Security

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Recommended Courses for M.Sc. Programme in Nuclear Security

Prerequisite courses

- NS.PR1. Applied mathematics
- NS.PR2. Basic nuclear physics

Required courses

- NS1. Introduction to nuclear security
- NS2. International and national legal framework regulating nuclear security
- NS3. Nuclear energy, nuclear fuel cycle and nuclear applications
- NS4 Methods and instruments for nuclear and other radioactive material measurements
- NS5. Effect of radiation, safety and radiation protection
- NS6. Threat Assessment
- NS7. Physical protection systems design and evaluation
- NS8. Physical protection technologies and equipment
- NS9. Security of nuclear and other radioactive material in transport
- NS10. Detection of criminal or unauthorized acts involving nuclear and other radioactive material out of regulatory control
- NS11. Interdiction of, and response to, criminal or unauthorized acts involving nuclear and other radioactive material
- NS12. Crime scenes investigation and forensic techniques

Recommended Courses for M.Sc. Programme in Nuclear Security Con't

Elective courses:

- NS13. Nuclear material accounting and inventory control of other radioactive material
- NS14. Vulnerability assessment of physical protection systems
- NS15. Risk assessment and management of State nuclear security measures
- NS16(a). Physical protection systems for nuclear and other radioactive material, sources and facilities
- NS16(b). Physical protection systems for radioactive material and sources
- NS17. Import/export and transit control mechanism and regime
- NS18. Nuclear security at major public events
- NS19. Nuclear forensics and attributions
- NS20. Infrastructure and procedures for detection and response to incidents involving nuclear or other radioactive material
- NS21. Cooperation of stakeholders at national and international level
- NS22. IT/Cyber-security

Experts holding a Nuclear Security M.Sc. Degree

Tactical Planning and Operational Activities

- Manage nuclear security at major facilities
- Analyse national nuclear infrastructure
- Develop strategy and arrange border control to detect and combat illicit trafficking
- Design physical protection system and evaluate its effectiveness
- Develop State's nuclear response plan and arrange respective response measures

Certificate Programme

- One semester programme: specialists with an overview in all areas of nuclear security and with specialization in particular topics
- Content: most M.Sc. courses reduced to contain only essential information
- Prerequisites and selection of additional courses are defined by respective universities or institutions
- Flexible programme to meet different national needs

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Specialists holding a Certificate in Nuclear Security

- Have a solid knowledge in all nuclear security areas
- Be able to prevent, detect malicious acts and respond to events involving nuclear and other radioactive material
- Effectively support and sustain an established nuclear security system

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Expressed Interest in Nuclear Security Education

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International Nuclear Security Network

IAEA Workshop, 29-31 March 2010

36 participants:
17 universities
2 international organizations
several other stakeholders

A partnership between the IAEA and universities, research institutions and other stakeholders has been established:

International Nuclear Security Education Network (INSEN)

Mission:
to enhance global nuclear security by developing, sharing and promoting excellence in nuclear security education

INSEN
Membership

<u>INSEN membership is informal and open to</u>:

- any educational and research institution already involved or, that plans to be involved in nuclear security education in the future.
- any other nuclear security stakeholder that is interested or involved in nuclear security education.

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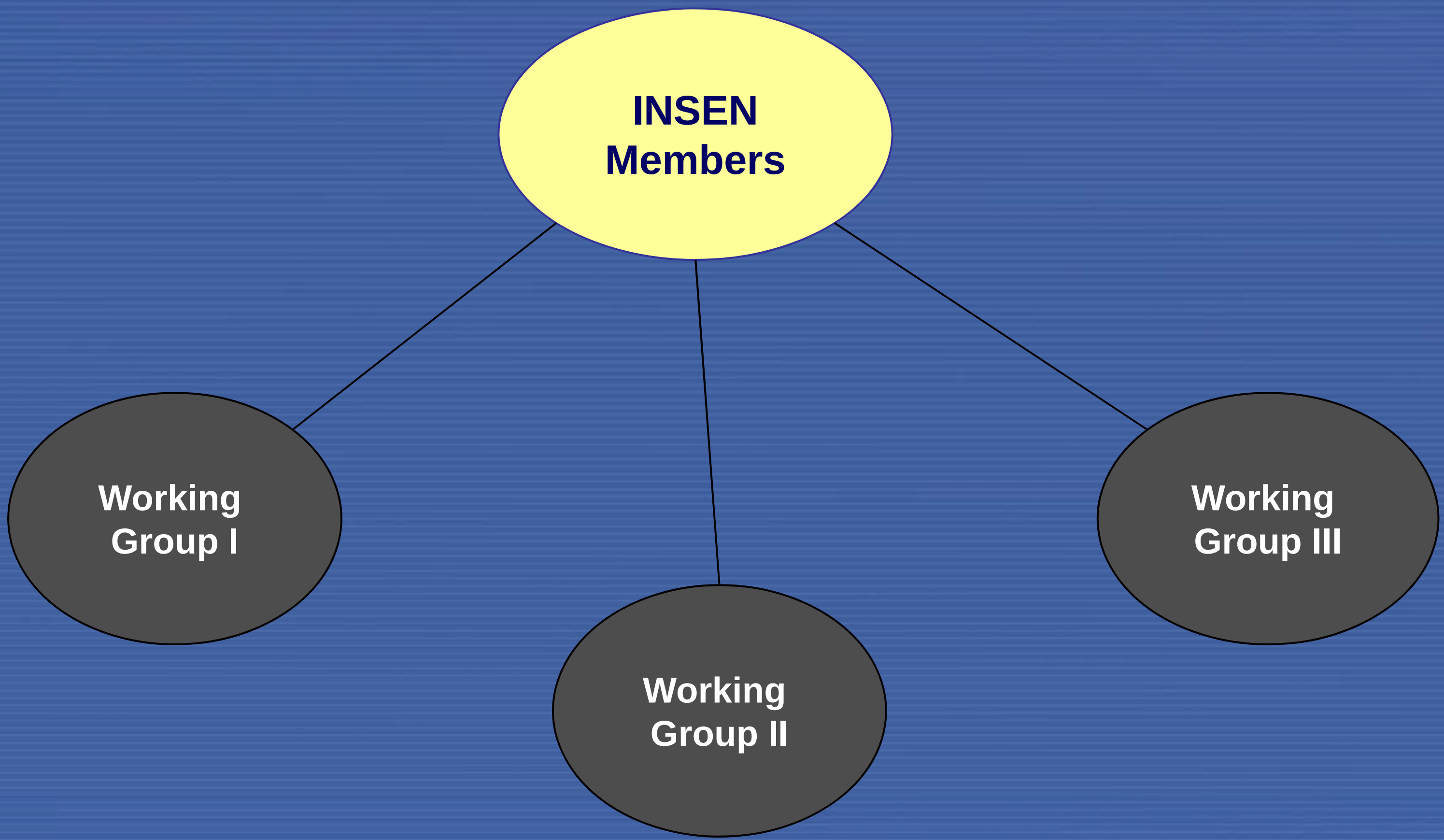
INSEN
Management & Structure



INSEN is guided by all members.

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INSEN
Working Groups

Working Group I:

Exchange of information and development of materials for nuclear security education

Working Group II:

Faculty development and cooperation among educational institutions

Working Group III:

Promotion of nuclear security education

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Conclusion

- Higher education in nuclear security plays an important role to assure international peace and security.
- First steps have already been taken to establish nuclear security education.
- International cooperation and collaboration is needed to accelerate the development of nuclear security experts.

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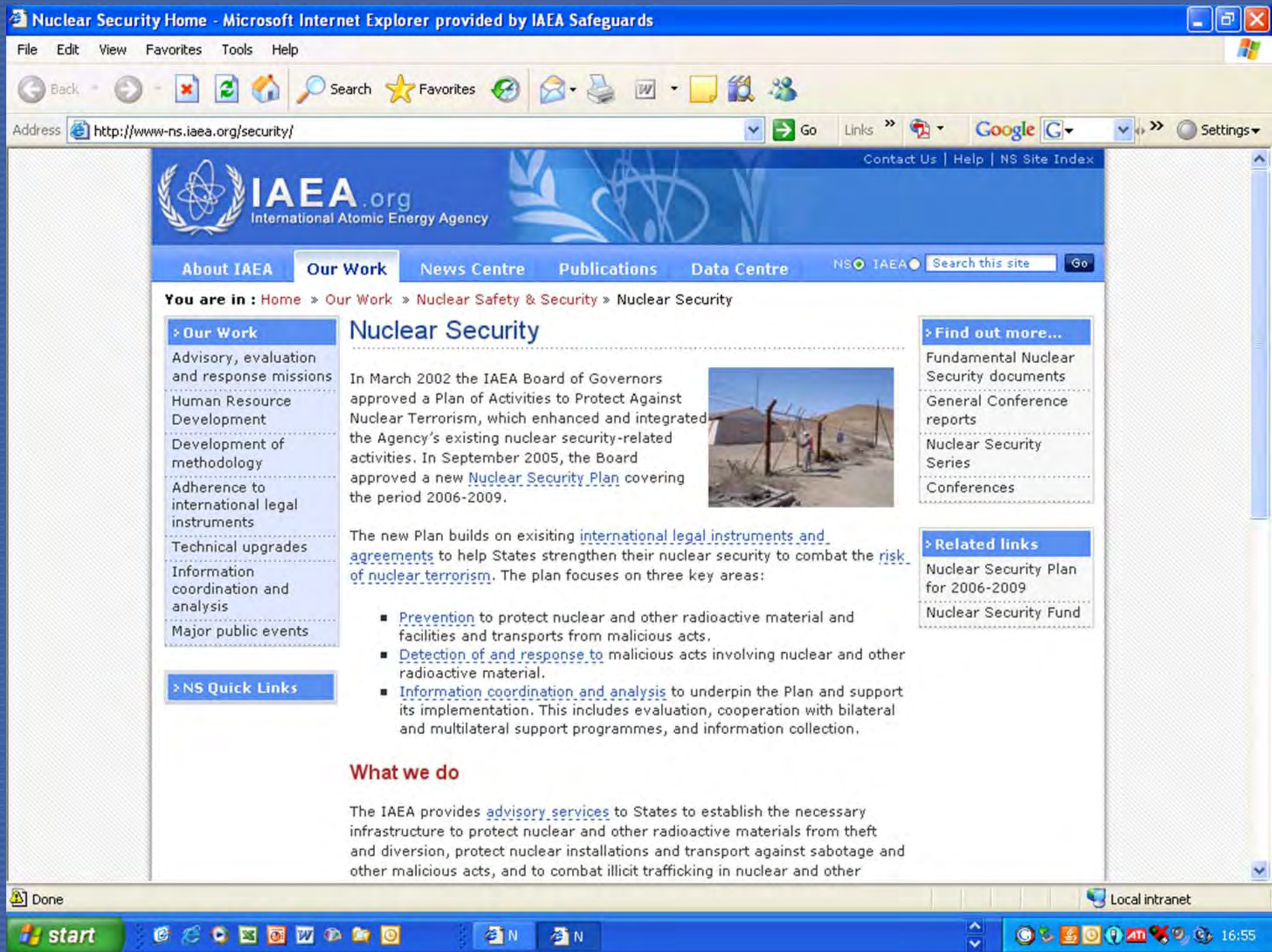
For further information please visit our website

http://www-ns.iaea.org/security/



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Eleonora Zakharko, Udo Weizel, Alexander Rduch

Use of Video Systems in Securing Nuclear Facilities

Outline of the presentation

- Principles of Application of Video Systems in Nuclear Facilities

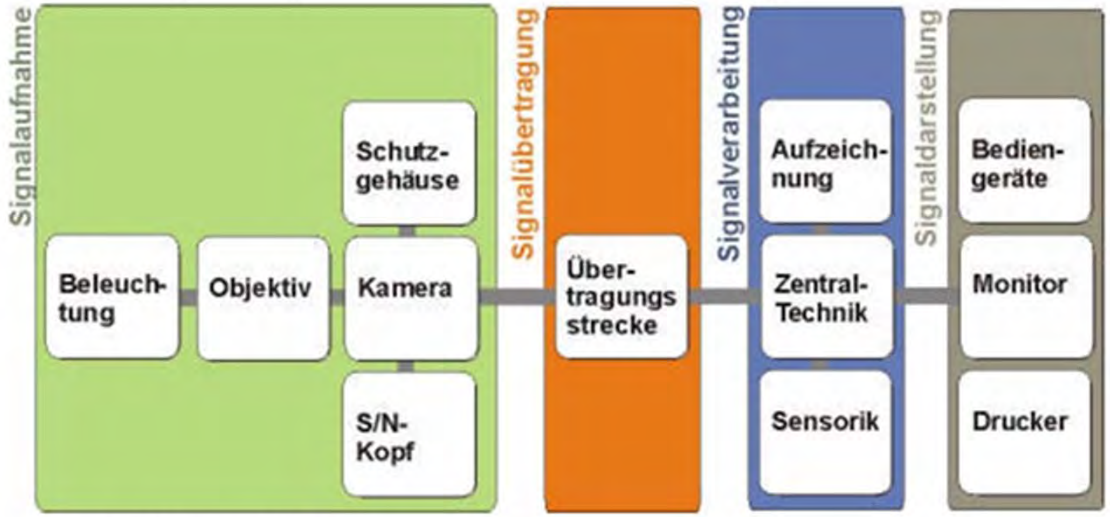
  Image obtaining → Signal transmission →

  Signal processing → Signal visualization

- Physical Protection Goes High Tech
  - Access Control
  - Video-Motion Detection
  - Management Systems
  - Closed-Circuit Television (CCTV): development of new principles of visual observation

Principles of Application of Video Systems in Nuclear Facilities

- CCTV - Closed Circuit Television as element of the physical protection of the nuclear facility
- Observation + verification of the alarms + detection
- General build-up of CCTV: image obtaining, signal transmission, signal processing and signal visualization



Legende: S/N - Schwenk-Neige

Image Obtaining: Requirements for Cameras

- Selection of camera type depends on specific requirements, boundary conditions, integration into the total (analog, IP cameras, resolution, black&white or coloured or respectively day-night mode depending on the illumination conditions)
- Number and positioning of cameras for complete surveillance or detection respectively, task depending, in each case optimal object capturing and perceptibility
- Task depending fixed or rotable mounting with rigid or zoomable lens, for detection possible cameras with fixed mounting and fixed lens adjustment

Image Obtaining: Requirements on Assembly of Cameras

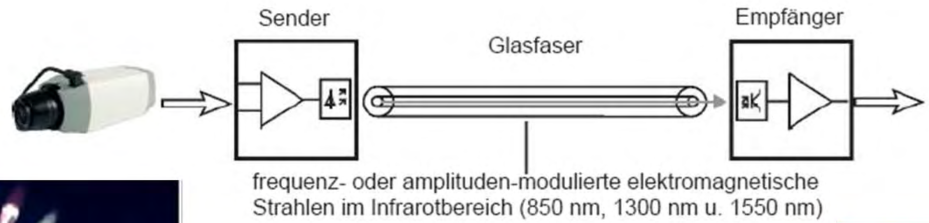
- Optimal positioning for sufficient image quality also in relation to natural and artificial light sources
  - Mounting with little vibration and oscillation, also under consideration of influences by the surroundings
- Suitable camera cases and covering of connections
  - Control of manipulation attempts
  - Protection against environmental impacts like dust, water, temperature, UV radiation, chemical and mechanical impacts, lightning etc. pp.

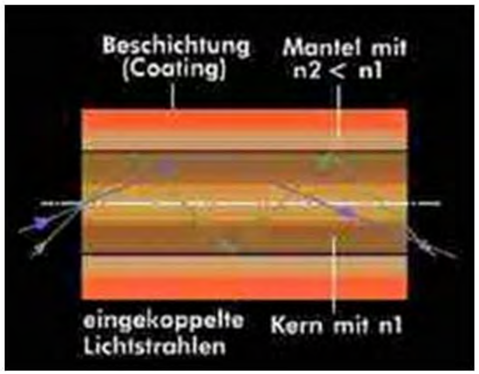
Image Obtaining: Aspects of Illumination for the Use of Video Systems

- Suitability of artificial light sources
- Highest colour fidelity and optimal efficiency of illumination construction
- Spectral energy distribution of light source in high agreement with sensitivity spectrum of camera

Signal Transmission: Requirements on Transmission of Video Signal

- Suitable transmission media are coaxial cable, twisted pare and glass fibre
- For use in nuclear facilities the glass fibre transmission is recommended





Signal Processing: Requirements on the Control System

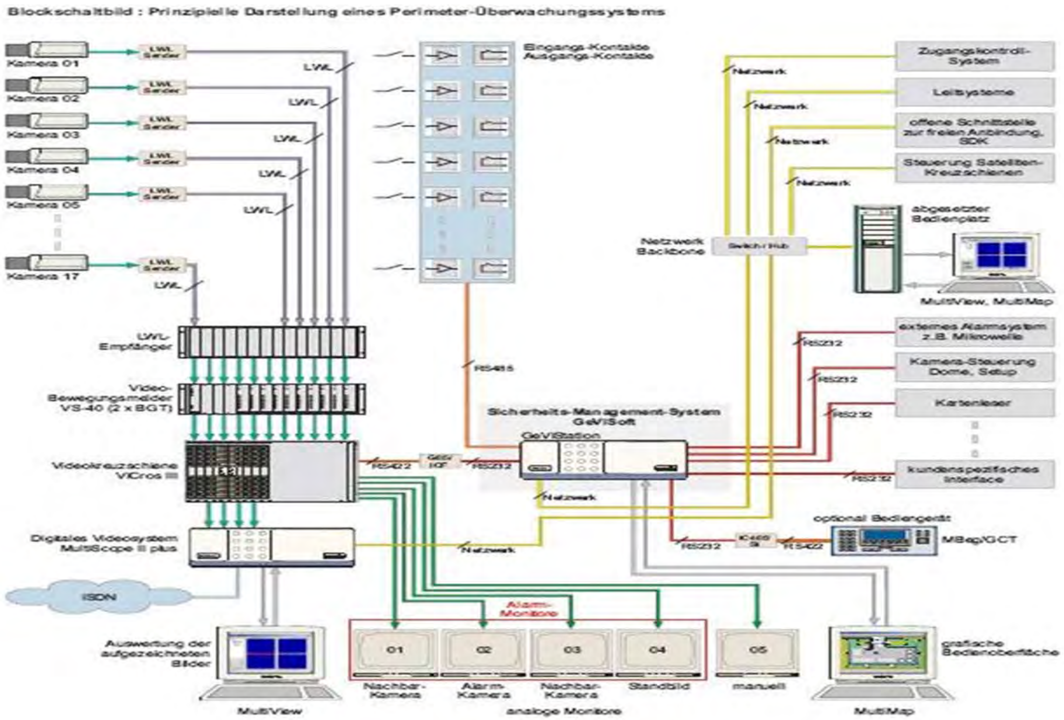
- Protection of central engineering against manipulations
  - protect central engineering with image recording physically if possible (i.e. by installing in separate system boards and rooms, possibly with limited access, locked and under control)
  - net based video technique also needs protection against IT criminals
- Possible compression methods with no losses or small compression

Signal Processing: Requirements on the Control System

- Image recording:
  - continuously for each camera with pre-determined image sequence and duration, with sufficient preliminary handling time and image quality for a query in case of alert
  - continuation in case of alert with higher image sequence, optimal for alert verification and query
  - recording of date und time for queries
  - colour recordings are preferable to black and white ones

Signal Processing: Combination with Systems of Danger Alarm Techniques

- Net based systems - possibly dedicated net for video system with defined interfaces for intersection with the separate nets of the associated systems



Signal Visualization: Display and Operation

- Ergonomical principles for the design of computer workstations
- Separate monitors for image switching in case of alert, monitors should possibly be black if there are no alert images on display

Requirements for Keeping the Functional Capability

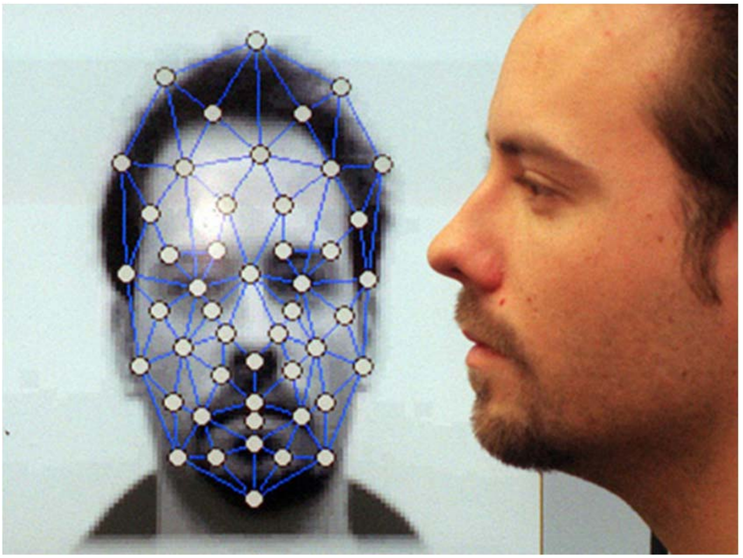
- Revisions, Inspections, Maintenance
  - periodical checks of functions, patrols
  - recurring examinations: al least annually, if necessary with participation of experts
  - periodical reviews of protection measures (deterministic security analysis): after several years in order to reflect the status on base of the state of the system of rules/requirements and on the state of technology and awareness
  - applicable alternative measures in case of failure of technical equipment

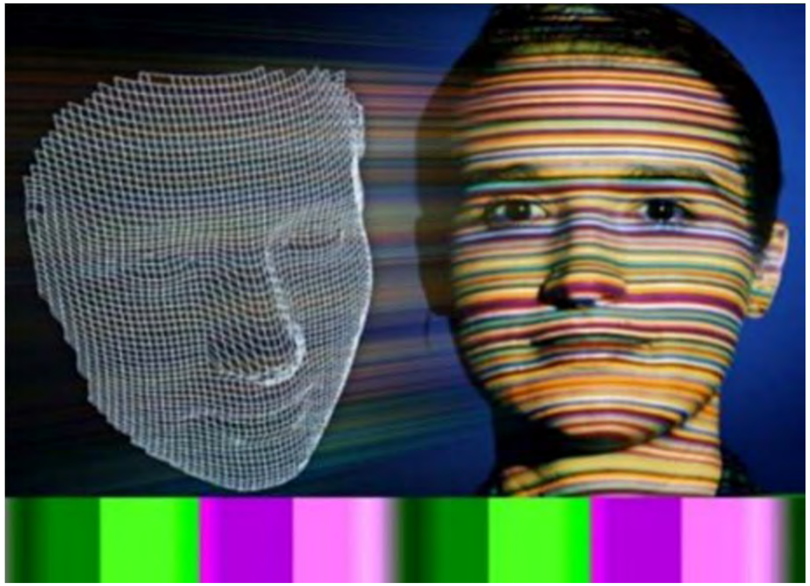
Physical Protection Goes High Tech

- Application areas of video systems and their innovations
  - Access Control
  - Video-Motion Detection
  - Management Systems
  - Closed-Circuit Television (CCTV): development of new principles of visual observation

Access Control

- Biometric identification systems
- Face Recognition, Fingerprint Image Processing, Iris Recognition, Retina Recognition, Hand Vascular Scanner, Palm Print

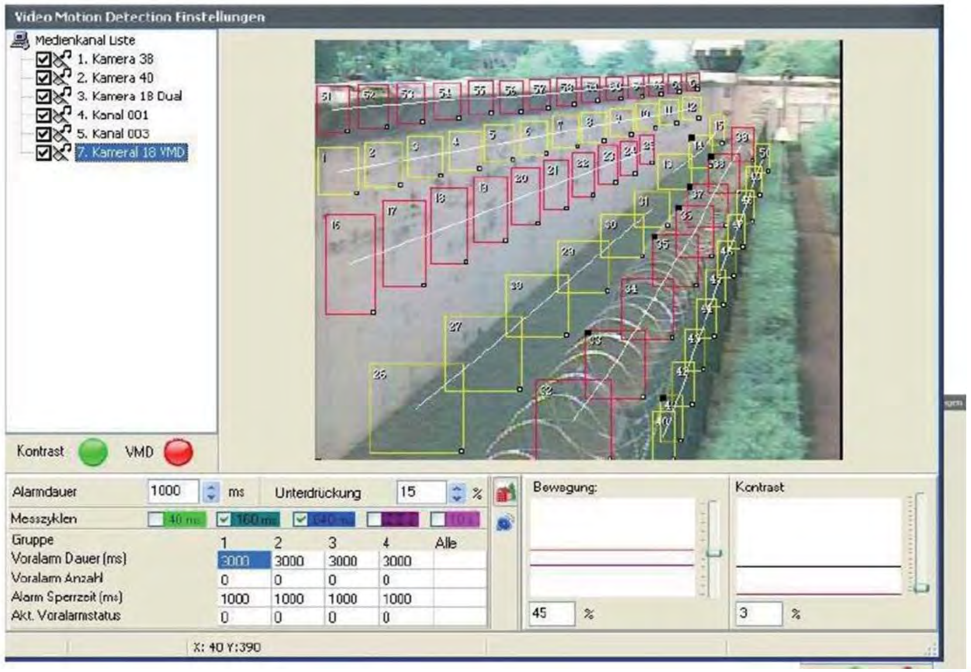




- 2D – Proceeding
- 3D – Proceeding

Video - Motion Detection

- Sensors – active/passive, Doppler effect, ultrasound
- Software: algorithm is based upon analysis of sensor fields, optimal positioning and parameterization as pre-condition for reliable detection



Management Systems

CCTV: Development of New Principles of Visual Observation

SEARISE - Smart Eyes: Attending and Recognizing Instances of Salient Events (FP7 EU Program, Objective Cognitive Systems and Robotics)
www.fit.fraunhofer.de