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Best Estimate Tools and Challenges of the New Reactor Designs

Tomislav Bajs
tomislav.bajs@enconet.hr
Enconet d.o.o., Zagreb, Croatia

Overview

- Introduction
- Characteristics of the New Reactor Designs
- Current Status of the Best Estimate (BE) Tools and Methodologies
- Specifics of Gen III Design Features on Safety Analysis
- Experience from the IAEA Review of the New Reactor Designs
- Concluding Remarks

Introduction

- BE tools (computer codes and methods) are widely used by the vendors, regulator and TSO's to provide the information on the safety margins of the new and operating reactors
- New reactor designs are developed to minimize potential risk to the public at overall competitive cost
- BE tools are incorporated in all the stages of the reactor design from preliminary/conceptual stage until licensing
- Such role places additional requirements on the approach to the accident analysis and the computational tools

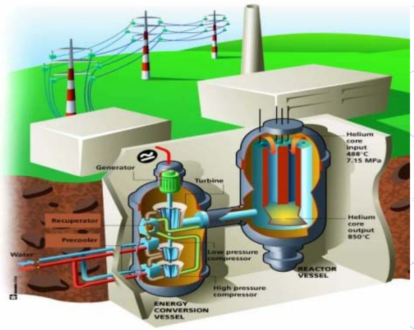
History of nuclear power

First reactors

Commercial power reactors

LWR with enhanced safety and performance

Fast reactors with closed fuel cycle



Generation I

- Shippingport
- Dresden
- Fermi I
- Magnox

Generation II

- LWR – PWR,BWR
- CANDU
- GCR
- VVER 440, 1000
- RBMK

Generation III

ABWR, EPR, AP-1000, VVER-1200 (AES 2006), APWR 1700, APR-1400

Generation IV

Comparison Generation II vs III

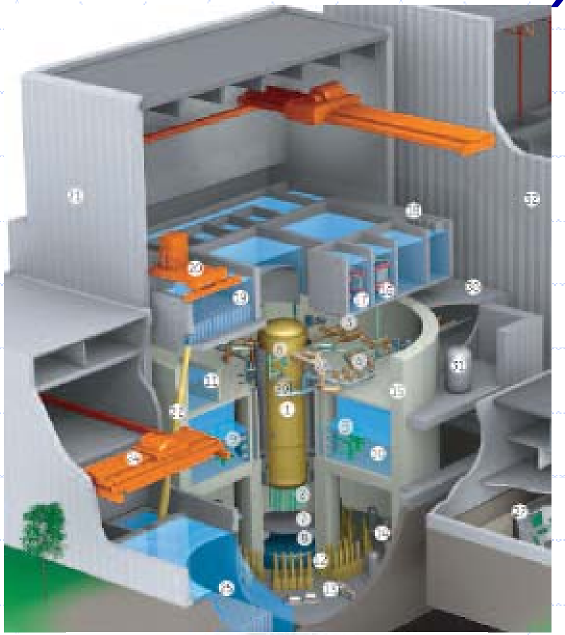
Gen II	**Gen III**
Power level up to 1000 MWe	Power level from 1100 to 1700 MWe,
Availability ~ 75-80%, efficiency ~ 30 %	Availability ~ 95%, efficiency up to ~ 40%
Base load operation	Load follow capability,
Plant life time 30-40 years	Plant life time (60-80 years)
CDF less than 1E-4 years LERF less than 1E-5 years	CDF less than 1E-5 ~ 1E-6 LERF ~ 1E-9 – 1E-6/year
Safety systems designed to cope with a set of DBAs	Systems for mitigation of severe accidents
Limited use of passive systems	Extended use of passive systems for some designs

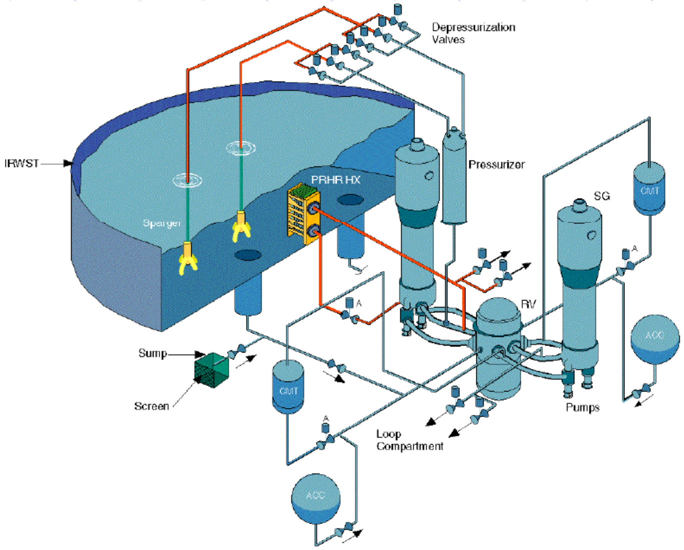
Comparison Generation II vs III (cont.)

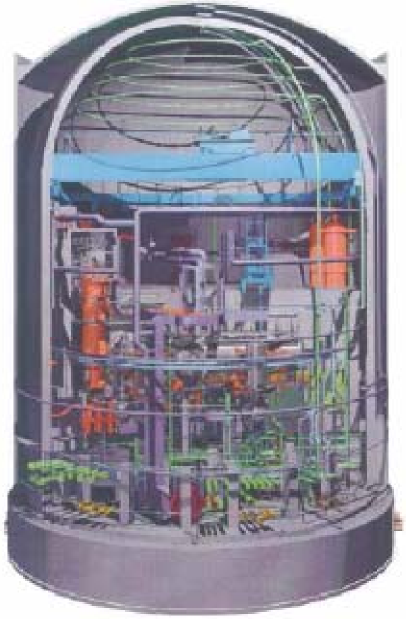
Gen II	**Gen III**
Severe accidents dealt with by means of accident management programmes (lack of dedicated systems)	Built-in systems and automatic actions combined with accident management programmes
Operator grace time minimum 30 minutes	Increased period without operator actions
Burn-up 30-40 MWd/kg	Higher burn-up to 60-70 MWd/kg in long term up to 100 MWd/kg)
Fuel cycle 1 year, extended to 18 months	Fuel cycle up to 24 months
	Robust double containment (with annulus venting), increased strength, designed against aircraft crash
	Seismic resistance of standard design 0.25 – 0.3 g

Some (Candidates for) Future Builds

- ABWR
- APWR
- ACR-1000
- AP1000
- APR-1400 (APR-1000)
- ESBWR
- VVER-1200 (AES 2006)







Overview of Codes - Background

- In general, the following two types of codes are used for the nuclear deterministic safety analysis:
  - **Mechanistic codes** include phenomenological models, necessary to provide an accurate prediction of the behaviour of an NPP (therefore also reffered to as best-estimate codes).
  - **Parametric codes** are a combination of phenomenological and user defined parametric models, necessary to describe the important trends in the behaviour of an NPP (historically, parametric codes have been developed first).

DBA Analysis Codes

- The type of code used for DBA analyses is decided by the component or system being analysed; and in general can be grouped into the following six categories:
  - Reactor physics codes
  - Fuel behaviour codes
  - Thermohydraulic codes, including system codes, subchannel codes, porous media codes, and Computational Fluid Dynamics (CFD) codes
  - Containment analysis codes
  - Atmospheric dispersion analysis and dose estimation codes
  - Structural analysis codes.

BDBA Analysis Codes

- Codes for the analysis of BDBAs can be categorized depending on the intended application, the level of modeling detail, the type of system considered and the phenomena addressed:
  - **Mechanistic codes**
    - for system thermohydraulics, progression of core damage and behaviour of fission products
    - for containment thermohydraulics, progression of damage and behaviour of fission products
    - separate effects codes for the analysis of separate processes such as steam explosions.
  - **Parametric codes** for system response and containment response.

Coupled Codes

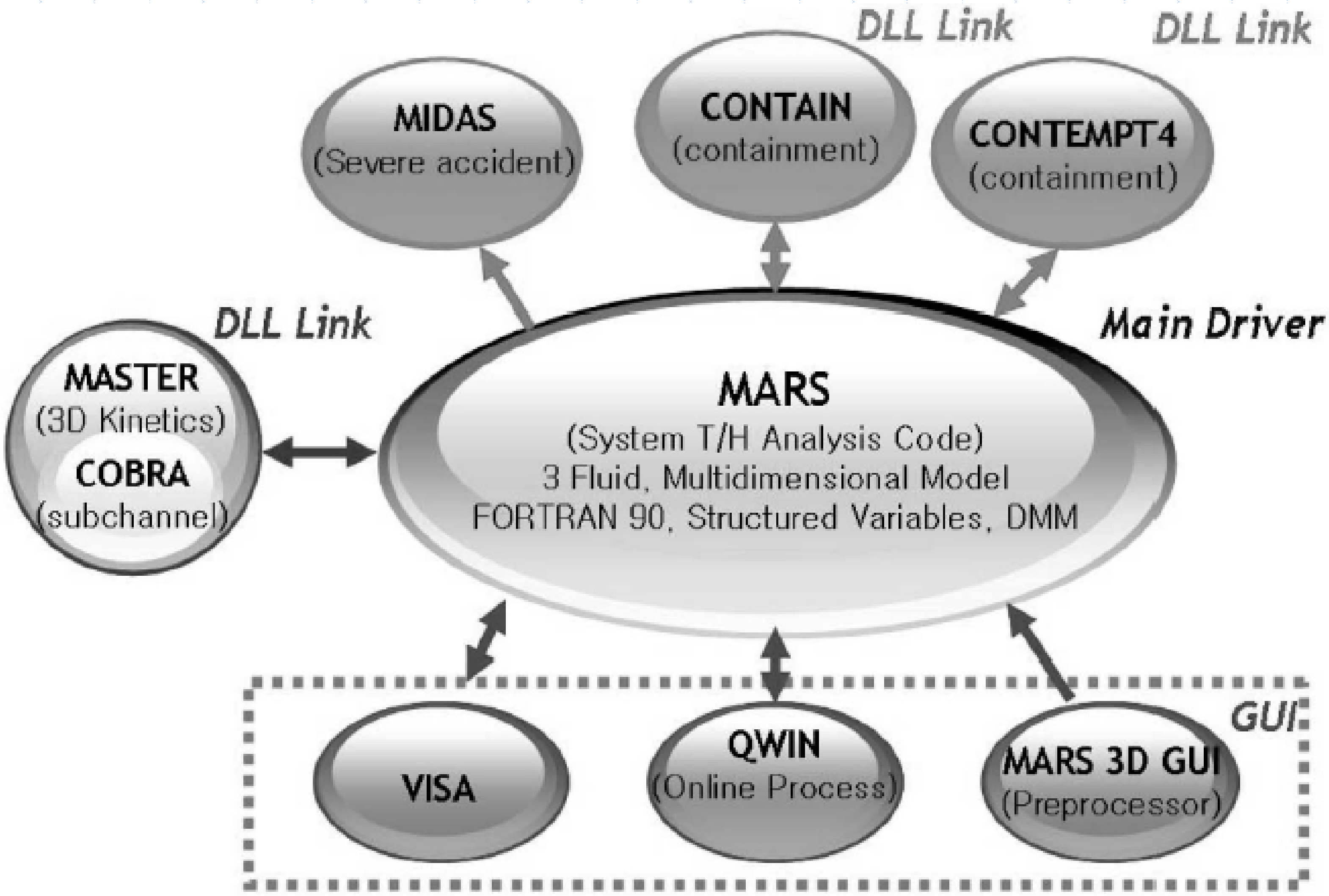
These codes are used for the following:

- Complex problems involving solution of thermal-hydraulics with other disciplines such as:
  - Reactor physics: neutron diffusion/transport
  - Fuel mechanics
  - Structural mechanics
  - Multidimensional flow
  - Chemistry, aerosol dynamics, metallurgy (severe accidents)

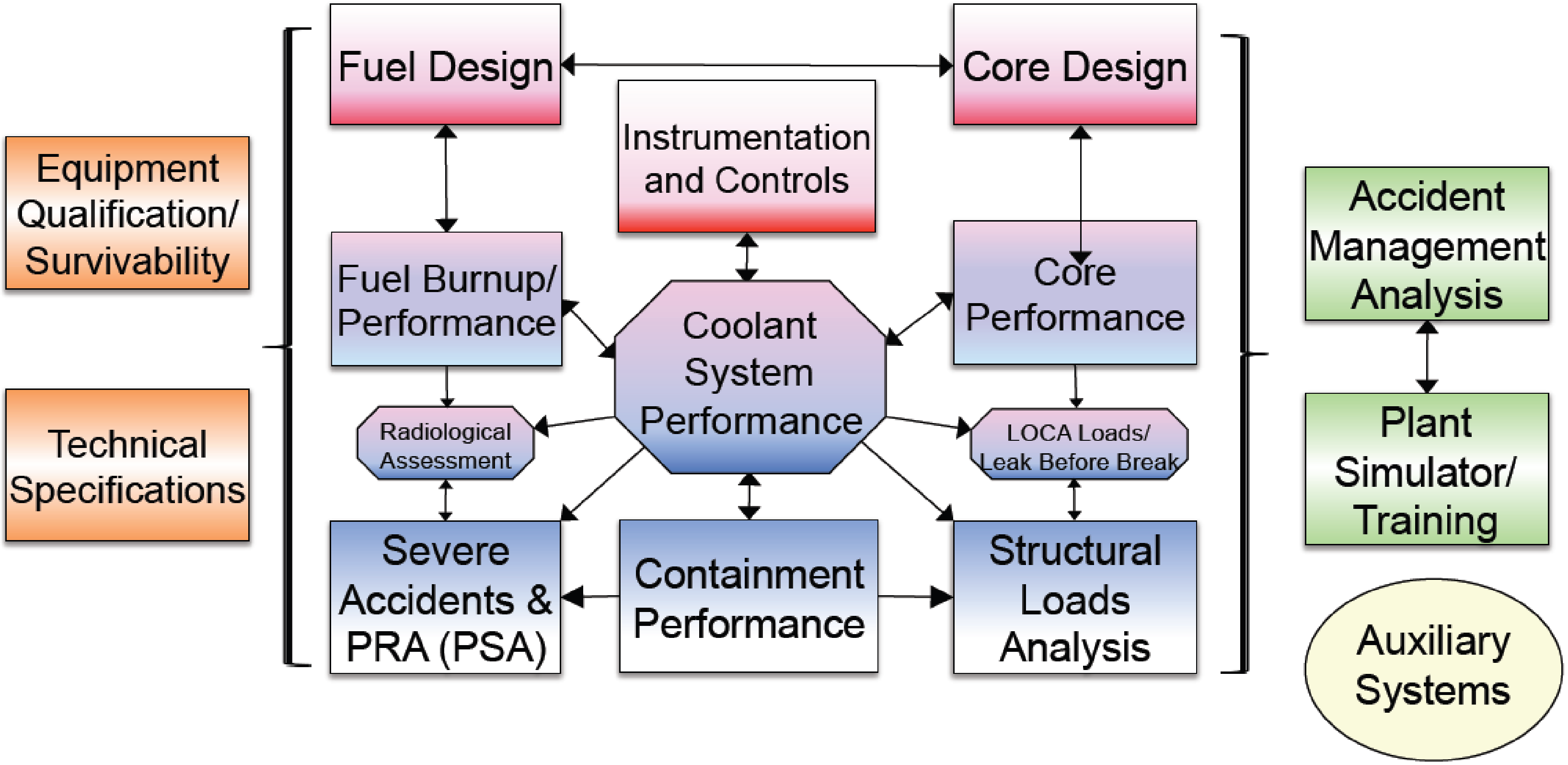
OR

- Interaction of domains (primary system – containment) with different physical models, spatial or time discretizations.

Integrated Methodology based on Coupled Code



Scope of NPP Analysis



Equipment Qualification/ Survivability

Technical Specifications

Fuel Design

Core Design

Instrumentation and Controls

Fuel Burnup/ Performance

Coolant System Performance

Core Performance

Radiological Assessment

LOCA Loads/ Leak Before Break

Severe Accidents & PRA (PSA)

Containment Performance

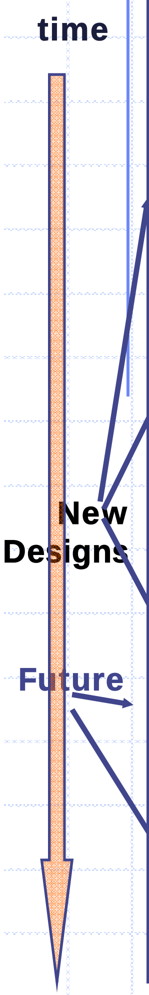
Structural Loads Analysis

Accident Management Analysis

Plant Simulator/ Training

Auxiliary Systems

Safety Analysis Methodologies



Selection of initiating events	Computer codes	Initial and boundary conditions	Plant systems availability	Evaluation of compliance with acceptance criteria
Few bounding cases (maximum credible accident)	Pessimistic models and codes	Pessimistic	Pessimistic	Some of calculated parameters compared to a set of limiting values
Spectrum of categorized events	Pessimistic (EM)	Pessimistic	Pessimistic	Graded criteria according frequency of initiating events
Spectrum of DBAs and BDBAs (incl. severe accidents)	BE models and codes	Pessimistic	Pessimistic with some relaxation based on probabilities (LOOP, ATWS, BDBAs)	Graded criteria according frequency of initiating events; for BDBAs, probabilistic criteria used
Spectrum of transients, DBAs and BDBAs (incl. severe accidents)	BE codes with uncertainties	Realistic data, with uncertainties	Pessimistic with some relaxation based on probabilities (LOOP, ATWS, BDBAs)	Uncertainty bands compared to a set of limiting values: graded criteria according frequency of initiating events; for BDBAs, probabilistic criteria typically used
Spectrum of transients, DBAs and BDBAs derived from their frequency	BE codes with uncertainties	Realistic data, with uncertainties	Availability of all systems in accordance with their failure probabilities	Calculated parameters (loads) with their probability distribution compared to a set of limiting values
Spectrum of transients, DBAs and BDBAs derived from their frequency	BE codes with uncertainties	Realistic data, with uncertainties	Availability of all systems in accordance with their failure probabilities	Calculated parameters (loads) with their probability distribution compared to a set of strengths with their distribution

Specifics of Gen III Design Features on Safety Analysis

Use of passive systems – low driving forces (gravitational) → more detailed modelling necessary (two-phase flow!)

Increased dimensions of the main components → importance of 3-D effects, revisiting scaling of results from experiments on the plant (calculation), additional validation of the codes

Large core dimensions → multidimensional neutronic and thermal hydraulic space effects (coupled-code applications)

Complex phenomena and dependence of response between different systems (primary, secondary, ECCS, containment) → control of transfer of information between the codes, validation of coupled codes

Load follow operation → modelling of control systems, consideration of effects on the core (localized burn-up effects)

Specifics of Gen III Design Features on Safety Analysis (cont.)

High thermal power with flat power profile → vulnerability of fuel assemblies requires more exact prediction of failed fuel rods and associated source term

New material (MOX!), geometrical, neutronic and thermal-hydraulic properties of fuel → reconsideration (including experiments) and introduction of revised models in the core

Fuel burn-up increase, use of burnable absorbers, longer core cycles → more detailed modelling of fuel behaviour in steady-state, transients and accidents

Specifics of Gen III Design Features on Safety Analysis (cont.)

Production, distribution, combustion and detonation of hydrogen in severe accidents is spatially dependent processes with potential for localized effects → detailed models for production, distribution and management of hydrogen needed

Severe accident management strategies (vessel retention, core catcher, PAR) → development of multidimensional models, experimental database, validation

Radiological acceptance criteria for operational states and for accidents, including severe accidents → elimination of unnecessary conservatism

Generic Reactor Safety Review (GRSR)

- Early evaluation of new reactor design safety cases based on Member States application.
- Review of technical documents for particular reactor design and its safety features based strictly on application of IAEA Standards:
  GSR Part 4 and NS-R-1.
- Harmonized review of safety cases made by vendors suitable for an individual evaluation or the licensing process.
- Flexible process applicable to mature designs as well as to concepts.

Generic Reactor Safety Review (GRSR)

UK HSE	Screening of Four New Reactor Safety Cases submitted for the consideration of the UK Health and Safety Executive/NII against DS348: ACR1000, AP1000, ESBWR, EPR
ATMEA	Screening of Conceptual Design Safety File and its innovative features against DS348 and NS-R-1 of new AREVA-MHI Reactor ATMEA1
AP1000	Screening of AP1000 Safety and Environmental Report and its innovative features against DS348 and NS-R-1
APR1400	Screening of KHNP APR1400 Safety and Environmental Report against DS348 and NS-R-1
APR 1000	Screening of KHNP APR1000 Safety and Environmental Report against GSR 4 and NS-R-1

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Generic Reactor Safety Review (GRSR) Outcomes

- Deterministic and probabilistic analysis were used as two complementary methods.
- Use of conservative computer codes for safety analysis for design basis accidents, the quantification of uncertainties in safety analysis only in selected applications (LOCA, DNBR)
- Application of older methods for high burn-up issues may be inappropriate
- More research is needed to the modelling of passive systems and the associated assumptions
- Additional selection and categorization of initiating events associated with passive safety systems failures needed
- In-vessel retention strategy for molten corium stabilization, passive containment cooling or by providing molten spread capability require additional supporting evidence.

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Observations on the Application of BE Tools

- Difficult to demonstrate that use of BE code in combination with conservative inputs and assumptions is done in a correct way:
  - Sensitivity calculations should confirm conservative selection of inputs (possible error if based on engineering judgment),
  - Intentional conservatisms may not always lead to conservative results since it can change during a course of the event, and may not be valid throughout the whole transient,
  - Conservatism may generate misleading sequences of events and unrealistic time-scales.
- Various aspects of the same initiating event when calculated in several steps by different computer codes:
  - Traceability of analyses questonable due to a lack of explanation how transfer of data between different steps is done
  - Transfer of data may mask some important phenomena.

Observations on the Application of BE Tools (cont.)

- Approach to severe accident analysis is not harmonized. Varies from predominantly probabilistic approach used in USA to the concept of reference severe accidents with deterministic criteria typical for Europe.

- Lack of information on survivability of systems in case of severe accidents, especially in cases complete loss of normal and emergency power supply. Acceptability of the design should be demonstrated using only systems dedicated to severe accident mitigation.

Examples of Deficiencies

- Omission of certain initiating events (usually accidents at shutdown operational modes or accidents in radwaste treatment systems or spent fuel management systems), or insufficient justification of selection or categorization of postulated initiating events
- State-of-the-art approaches and widely used international practices not followed, such as use of best estimate codes
- Inadequate demonstration of codes validation or use of computer codes beyond the range of their validity (e.g. heat transfer correlations)
- Inadequate selection of acceptance criteria for high burn-up fuel

Examples of Deficiencies (cont.)

- Sensitivity analyses not provided for selection of a bounding case or not adequately convincing
- Missing data important for evaluation of radiological status prior the accident (cladding defects, excessive coolant radioactivity, and leaking steam generator tubes)
- Assumptions used in safety analysis not presented in a clear and convincing way
- Inconsistencies in transfer of data (without sufficient justification) from thermal-hydraulic analysis to containment analysis and to source term analysis
- Unexpected rapid increase of doses in the environment with decreasing probability of occurrence in the range 1E-6 – 1.E-7/r.year
- Unclear application of the single failure criterion

Examples of Deficiencies (cont.)

- Over-conservatism used in analysis of design basis accidents (e.g. postulation of a core melt) leading to the conclusion that radiological consequences of design basis accidents are more severe than of severe accidents
- An explicit list of transients and accidents occurring during shutdown operational regimes not provided
- Missing analysis of plant normal operation conditions
- Limited documentation of computer codes and demonstration of their validation status
- Limited presentation of plant data used in accident analysis (including reference values and uncertainties of plant parameters, set-points and system characteristics)

Examples of Deficiencies (cont.)

- Missing uncertainty and sensitivity calculations to prove adequate selection of conservative inputs, missing reference to such calculations
- Termination of analysis prior to reaching safe stable status of the plant
- Interpretation and evaluation of results made beyond the scope of performed analysis, in particular in cases when several acceptance criteria apply for the same event.
- Inconsistencies in use of plant data in analysis performed by different computer codes and in transfer of data between various stages of analysis.

Conclusions

- New Generation III reactors are significantly improvement in safety as compared to Generation II.
- Majority of design features of Generation III reactors are evolutionary using proven technologies, but there are significant challenges that require careful consideration during safety assessment.
- BE safety analysis and quantification of uncertainties, should be more broadly used in demonstration of safety margins for new plant designs.
- Use of coupled codes with appropriate methodologies and validation might overcome limitations of the multy code-multi step approach that is still used in licensing of new designs.

Acknowledgments

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Thank you for your attention!