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L9.2 Safety and Regulatory Considerations in the Bid Specifications and Bid Evaluation

Jozef Mišák, Director for Strategy
Nuclear Research Institute Rez plc

IAEA/ANL Regional Workshop on Establishing a Nuclear Safety Infrastructure for a National Nuclear Power Programme
29 November - 10 December 2010, ANL, USA
NRI Rez - EXAMPLES OF ACTIVITIES

Activities:
- Energy R&D
- Design of power plants
- Engineering services
- Treatment of radwaste
- Radiopharmaceuticals
- Industrial applications
- TSO for Czech nuclear regulatory body

Comparison experiment vs FLUENT analysis (PANDA facility)
Display of NPP Temelín MCR
Qualification of NPP cables
Operation of Pb-Bi loop
Lay-out of a deep geological repository
Digitalised as-built NPP Temelín
Temperature field at R outlet at MCP start-up
Dynamic analyses of in pipes
PET Centre hospital
Demonstration bitumenation unit
Welding for spent fuel repacking
Nuclear facilities in the Czech Republic
COUNTRY PROFILE
RESEARCH REACTORS

NRI Řež Plc.
LVR-15
(max. power 10 MWt)

NRI Řež Plc.
LVR-0 – zero power research reactor, modified fuel of WWER – 440, 1000 type; measurements of basic physical fuel parameters

Czech Technical University, Prague
VR – 1 (zero power, 19,7% 235U)
NPP Dukovany in operation

- VVER 440 - 213, 4 units
- PWR, 6 loops, 2 turbines
- Pressure-suppression containment
- 1360 MWt, 440 MWe
- In operation since 1985 - 1987

→ dry interim storage of spent fuel (cask-type, CASTOR),
→ regional shallow land repository of radioactive waste to accommodate all low and intermediate radioactive wastes from both nuclear power plants
VVER 1000 – 320, 2 units
PWR, 4 loops, 1 turbine
Full-pressure containment
3000 MWt, 1000 MWe
Construction since 1986
Operation since 2003-2004

dry interim storage of spent fuel (cask-type, CASTOR) under construction
RADWASTE AND SPENT FUEL STORAGE FACILITIES (DUKOVANY SITE)

SHALLOW LAND REPOSITORY

INTERIM SPENT FUEL STORAGE
Construction of new units: Current Status

- Process to build 2 new units at Temelin since July 2008
  - EIA Study Completed, currently reviewed by the authorities
  - Invitation for Tender issued by the utility (August 2009)
  - Pre-qualification of bidders completed (January 2010), tender ongoing: Areva, Atomstroyexport, Westinghouse
  - Initial Safety Analysis Report in progress (site approval stage)

- Plans to build 1 new unit at Dukovany site:
  - Development of a Feasibility Study for construction of Unit 5 at Dukovany site has started
  - Areva, Westinghouse, ASE, MHI, KOPEC and ATMEA considered
Content of the presentation

- Overall structure of development of a nuclear project and its licensing in the CR
- Legislative system in the Czech Republic
- Objectives of the Feasibility Study and of Bid Specification and evaluation
- Documents to be developed by the utility
- Use of European Utility Requirements for specification of safety requirements
- WENRA Reference Levels
- Selected technical issues to be considered more carefully in utilizing EUR for development of Bid Invitation Specification (examples)
- Conclusions
Estimated time for construction of a new unit is ~13 years!
Legislative system in the Czech Republic is well developed

The basic legal instruments governing the licensing and approval process for nuclear installations are:
- Civil Construction Act (No. 186/2006 Coll.);
- Atomic Act (No. 18/1997 Coll.) and related Decrees (e.g. on QA, Design, Operation);

Other important legal instruments in this area are:
- Administrative Procedure Act (No. 500/2004 Coll.);
- State Inspection Act (No. 552/1991 Coll.);
- Environmental Impact Assessment Act (No. 100/2001 Coll.)
LICENSING IN CONNECTION WITH NEW NPPs

There are two kind of permits to be dealt with: the local permits and the nuclear permits:

- **Local Permits** include the preparation of the information and documents required by local administration to start the works. One of the main documents within this package is the Environmental Impact Assessment (EIA), which must be approved in Public Hearing before starting the conditioning at site.

- **Nuclear Permits.** These should be granted for the starting of Construction Activities and for the procurement of the equipment comprising the NI, as well as other safety related equipment.
  - **Initial Safety Analysis Report**, which is the basis for site approval
  - **Preliminary Safety Analysis Report** (PSAR) and its approval, is a prerequisite to start Construction; and the
  - **Final Safety Analysis Report** (FSAR) and its review, to load nuclear fuel and start the NPP tests and commissioning.
Process of licensing
Feasibility study

- **Purpose:** Set up conditions necessary for new NPP construction
  - Technical feasibility (safety, environmentalism, Temelin site)
  - Economical feasibility
  - Legal feasibility (Compliance with Czech and international legislation)

- **Feasibility study content:**
  - Technical Evaluation
  - Site Information
  - Contract Approaches
  - Project Management
  - Schedule
  - Financing
  - Economic Analysis
  - Final Conclusions
Objectives of the Feasibility Study and of Bid Specification and evaluation

- To demonstrate the technical and economical feasibility of the project,
- To identify the conditions needed to develop the project and to outline the actions necessary to address them
- More specifically:
  - Confirmation that the design is in compliance with the relevant regulations of the country,
  - Confirmation that the plant can be built on a given site, taking into account site specific characteristics, available construction space,
  - Confirmation that all large and heavy components can be transported to the site at acceptable cost and time duration,
Objectives of the feasibility study (cont’d)

- Determination of basic limiting conditions for optimal utilization of the given site,
- Confirmation that the design under consideration has features which are adequate for obtaining public acceptance and political support (including EU level),
- Confirmation that the design can be safely operated in the Czech electrical grid, with expected operational characteristics
- Confirmation that there are sufficient margins in the design in order to allow for operational flexibility and future potential changes in safety requirements
- Confirmation that the Vendor has adequate resources and experience to manage the project under consideration with high quality, in mutually agreed time schedule and cost
Objectives of the feasibility study (cont’d)

- Confirmation that there is very low risk in delayed construction and unexpected problems in reliable operation
- Confirmation that there is insignificant financial risk associated with the plant construction and operation
- Confirmation that the design is reflecting the current state of the art of nuclear technologies,
- Confirmation that the design can be built and operated from viewpoint of the available either current or reasonably accessible manpower and financial resources
- Identification of problems potentially requiring design modifications.
Feasibility study

"FEASIBILITY STUDY OF THE NEW NUCLEAR POWER PLANT IN THE CR"

LOKALITA TEMELÍN

KAP 2 SITE INFORMATION

ČESKÁ VERZE

INVESTOR: ČEZ, a.s., Duhová 21444 149 50 Fráňa 4

Datum: 07/2007
Feasibility study

• About 100 engineers involved in the development
• In addition to utility, 3 other organizations contracted
• About 1 year of work
Volume of information in response of Bid Invitation Specification according EUR (SKODA Alliance offer for construction of Belene NPP)
Temelin 3,4 supply model

- Nuclear Island
  - Nuclear Fuel
  - I & C, Electrical Systems, Building
  - BOP Balance of Plant

- Conventional Island

- Power Island

- Public contract EPC

- Power Plant

- Related Investments

- Related investments of other investors
How to specify safety requirements in Bid Invitation Specification for new NPPs
Documents to be developed by the utility

- Bid invitation specification
- Environmental Impact Assessment
- Initial Safety Analysis Report

- All these documents should be consistent
- Quite complicated to achieve, in particular if the reactor type is not selected yet
Background documents

- European Utility Requirements for LWR Nuclear Power Plants. Revision C, April 2001
- Set of national legislative documents
- Reactor Harmonization Group, WENRA Reactor Safety Reference Levels, January 2008
- WENRA Statement on safety objectives for new nuclear power plants, November 2010
- Other IAEA Safety Standards
European Utility Requirements

Main objective: A common set of utility requirements, endorsed by the major European electricity producers for the next generation of LWR nuclear power plants

Development of Standard Designs to be built and licensed in EU with only minor variations

Around 5000 requirements formulated

The utility requirements address the designers and suppliers of LWR plants, with the aim to promote harmonization of:

- safety approaches, targets, criteria and assessment methods
- standardization of design conditions
- design objectives and criteria for the main systems and equipment
- equipment specification and standards
- information required for safety, reliability and cost assessment

Volumes 1.3 and 2.1 of EUR include detail safety requirements (more than 150 pages) which can be used as a starting point for formulation of safety requirements in the Bid Invitation Specification.
Use of EUR for specification of safety requirements

1.3 1 OBJECTIVES OF THE EUR SAFETY REQUIREMENTS
1.3 2 LICENSABILITY
1.3 3 SAFETY APPROACH
1.3 3.1 Deterministic and probabilistic approaches
1.3 3.2 Design Basis Conditions
1.3 3.3 Design Extension Conditions
1.3 3.4 Internal and external hazards
1.3 4 QUANTITATIVE SAFETY OBJECTIVES
1.3 4.1 Probabilistic targets
1.3 4.2 Off-site release limits during Normal Operation and Incidents
1.3 4.3 Off-site release targets for Accidents
1.3 4.4 Release targets for Severe Accidents
1.3 5 SAFETY CLASSIFICATION OF FUNCTIONS AND EQUIPMENT
Use of EUR for specification of safety requirements

2.1 1 FUNDAMENTAL SAFETY OBJECTIVES AND POLICIES
2.1 1.1 Fundamental safety objectives
2.1 1.2 Safety policy
2.1 1.3 Defence in depth

2.1 2 QUANTITATIVE SAFETY OBJECTIVES
2.1 2.1 Overall approach to Targets
2.1 2.2 Radiological impact during Normal Operation and Incident Conditions
2.1 2.3 Operational staff doses during Normal Operation and Incident Conditions
2.1 2.4 Off-site release Targets for Accident Conditions
2.1 2.5 Off-site release Targets for Design Extension Conditions
2.1 2.6 Probabilistic safety Targets
2.1 2.7 Probabilistic safety assessment methodology

2.1 3 DESIGN BASIS CONDITIONS
2.1 3.1 Deterministic approach to safety
2.1 3.2 Design basis and safety objectives
2.1 3.3 Deterministic safety analysis
2.1 3.4 Single Failure Criterion

2.1 4 DESIGN EXTENSION CONDITIONS
2.1 4.1 Design extension approach
2.1 4.2 General assessment Rules for DEC
2.1 4.3 Complex Sequences
2.1 4.4 Severe Accidents
2.1 4.5 Severe Accident In-Containment Source Term quantification
Use of EUR for specification of safety requirements

2.1 4.5.1 General approach to the In-Containment Source Term

2.1 5 EXTERNAL AND INTERNAL HAZARDS

2.1 5.1 Hazards to be considered
2.1 5.2 Approach to hazards
2.1 5.3 External hazards
2.1 5.4 Internal hazards

2.1 6 ENGINEERING REQUIREMENTS

2.1 6.1 Design objectives
2.1 6.2 Design measures to achieve reliability of functions
2.1 6.3 Design Codes and Standards
2.1 6.4 Materials
2.1 6.5 Plant performances following Accident Conditions
2.1 6.6 Plant performances following Design Extension Conditions
2.1 6.7 Autonomy objectives
Use of EUR for specification of safety requirements

2.1 6.8 Classification of Safety Functions and categorisation of equipment
2.1 6.9 Equipment qualification
2.1 6.10 Inspection, on-line monitoring, testing and maintenance
2.1 6.11 Human factors
2.1 6.12 Main and emergency plant control
2.1 6.13 Accident management
2.1 6.14 Radiation protection
2.1 6.15 Quality Assurance

2.1 7 SITE CONDITIONS
2.1 7.1 Factors affecting choice of site
2.1 7.2 Hazards
2.1 7.3 Surrounding population
2.1 7.4 Reliability of services
Use of EUR for specification of safety requirements

2.1 8 TABLES

8.1 Table 1: Radiological criteria for radioactive releases in Normal Operation and Incident Conditions

8.2 Table 2: Frequencies and acceptance criteria for Normal Operation, Incident Conditions and Accident Conditions

8.3 Table 3: List of Design Basis Conditions

8.4 Table 4: Hazards

8.5 Table 5: Fuel acceptance criteria in Design Basis Category 4 Conditions

A SOURCE TERM AND RELEASE QUANTIFICATION METHODOLOGY FOR DESIGN EXTENSION

B VERIFICATION PROCESS OF THE EUR ENVIRONMENTAL IMPACT TARGETS
WENRA Reference Levels (01/2008)

- Reference Levels: a consensual opinion on safety of the WENRA regulators (17 European countries)
- A set of minimum requirements (Reference Levels) for harmonizing reactor safety in 18 areas for existing NPPs
- WENRA regulators: harmonization of safety level should not be a matter of voluntary decisions or agreements made with the nuclear industry, but the requirements should be implemented into national legislative basis of the countries involved and subsequently into operation of existing NPPs
- The agreed year for harmonization of legislation is 2010
## WENRA Reference Levels

**January 2008**

<table>
<thead>
<tr>
<th>Issue</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>A:</td>
<td>Safety Policy</td>
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<tr>
<td>B:</td>
<td>Operating Organisation</td>
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<tr>
<td>C:</td>
<td>Management System</td>
</tr>
<tr>
<td>D:</td>
<td>Training and Authorization of NPP staff</td>
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<tr>
<td>E:</td>
<td>Design Basis Envelope for Existing Reactors</td>
</tr>
<tr>
<td>F:</td>
<td>Design Extension of Existing Reactors</td>
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<tr>
<td>G:</td>
<td>Safety Classification of Structures, Systems and Components</td>
</tr>
<tr>
<td>H:</td>
<td>Operational Limits and Conditions</td>
</tr>
<tr>
<td>I:</td>
<td>Ageing Management</td>
</tr>
<tr>
<td>J:</td>
<td>System for Investigation of Events and Operational Experience Feedback</td>
</tr>
<tr>
<td>K:</td>
<td>Maintenance, In-service inspection and Functional Testing</td>
</tr>
<tr>
<td>LM:</td>
<td>Emergency Operating Procedures and Severe Accident Management Guidelines</td>
</tr>
<tr>
<td>N:</td>
<td>Contents and updating of Safety Analysis Report</td>
</tr>
<tr>
<td>O:</td>
<td>Probabilistic Safety Analysis</td>
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<tr>
<td>P:</td>
<td>Periodic Safety Review</td>
</tr>
<tr>
<td>Q:</td>
<td>Plant Modifications</td>
</tr>
<tr>
<td>R:</td>
<td>On-site Emergency Preparedness</td>
</tr>
<tr>
<td>S:</td>
<td>Protection against Internal Fires</td>
</tr>
</tbody>
</table>
Comparison of background documents

- WENRA Reference Levels represent reasonably balanced requirements applicable for both existing and new designs
- WENRA Levels have been derived from IAEA Safety Requirements, partially also from Safety Guides
- Good consistency between WENRA and IAEA
- In several cases wording of WENRA more stringent using statements such as “…shall exist”, “…shall be possible”, etc rather than “adequate consideration shall be given…” used by IAEA
- Compliance with WENRA also provides for compliance with the IAEA Safety Requirements
- In principle, WENRA is for existing reactors, EUR is for new reactors
Comparison of background documents

- EUR provides very useful and detailed guidance
- Level of details in EUR is even higher than IAEA Safety Guides and corresponds to other IAEA guidance documents (i.e. Safety Reports or TECDOCs)
- Much more details are included in the EUR, such as
  - Quantitative specification of deterministic and probabilistic targets (never done in IAEA Standards)
  - Specification of certain computational methods (hydrogen, containment loading, radiation doses, etc)
  - Prescription of some engineering solutions to address the challenges.
- No contradictions found between EUR and WENRA/IAEA, but differences in terminology and level of details
- Level of details in EUR in some cases unnecessarily prescriptive
How the issue was addressed in the Czech Republic?

- **Bid Invitation Specification**: based on the text of EUR with more detailed or different specification of selected issues

- **Environmental Impact Assessment Study**:  
  - Scope of the document specified in the legislation  
  - Bounding (enveloping) characteristics used taking into account information collected in the feasibility study  
  - No specific reactor type named in the documentation in order not to rank the vendors during the negotiation

- **Initial Safety Analysis Report**:  
  - Site characteristics described in accordance with the results of the site evaluation  
  - Description of future plant expressed in terms of combination of safety requirements  
  - In combination of safety requirements the relevant national legislation, the IAEA Safety Requirements (design, safety assessment, operation, etc) and WENRA Reference Levels used
Selected technical issues to be considered more carefully in utilizing EUR for development of Bid Invitation Specification (examples)
Consideration of Czech legislation

- ACTS
- REGULATIONS (DECREES)
- GUIDES
- INDUSTRIAL STANDARDS
Czech regulations applicable for NPPs

- Obligations can be imposed on the stakeholders exclusively by acts.
- Regulations (decrees) can specify in more details the obligations, no new obligations can be introduced.
- Acts and regulations are legally obligatory, to be followed verbatim.
- Guides are optional rules and explanations issued by the regulatory body; they are not obligatory, but if followed they facilitate licensing.
- Industrial standards in general are not obligatory, except several cases when prescribed by a regulation (radiation protection, fire safety, safety of labour, civil constructions).
- Standards of different origins (US, French, Russian, German, ...) are acceptable, use of selected standards shall be specified by the vendor, consistency of standards very important.
Czech regulations applicable for NPPs

- Act No. 18/1997 Coll. on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Act)
- Decree of the SÚJB No. 215/1997 on Criteria for Siting Nuclear Facilities and Very Significant Ionising Radiation Sources
- Decree of the SÚJB No. 195/1999 Coll. on Basic Design Criteria for Nuclear Installations with Respect to Nuclear Safety, Radiation Protection and Emergency Preparedness
- Decree of the SÚJB No. 195/1999 Coll. on Basic Design Criteria for Nuclear Installations with Respect to Nuclear Safety, Radiation Protection and Emergency Preparedness
- Decree of the SÚJB No. 106/1998 Coll. on Nuclear Safety and Radiation Protection Assurance during Commissioning and Operation of Nuclear Facilities
- Decree of the SÚJB No. 185/2003 Coll. on Decommissioning of Nuclear Installation or Category III. or IV. Workplace
- Decree of the SÚJB No. 144/1997 Coll. on Physical Protection of Nuclear Materials and Nuclear Facilities and their Classification, amended in Decree of the SÚJB No. 500/2005 Coll.
Czech regulations applicable for NPPs


- Decree of the SÚJB No. 132/2008 Coll. on Quality Assurance System in carrying out activities connected with utilization of nuclear energy and radiation protection and on Quality assurance of selected equipment in regard their assignment to classes of nuclear safety

- Decree of the SÚJB No. 307/2002 Coll. on Radiation Protection

- Decree of the SÚJB No. 317/2002 Coll. on Type Approval of Packaging Assemblies for Transport, Storage and Disposal of Nuclear Materials and Radioactive Substances, on Type Approval of Ionizing Radiation Sources and on Transport of Nuclear Materials and Specified Radioactive Substances (”on Type Approval and Transport”), amended in Decree SÚJB No. 77/2009 Coll.
Czech regulations applicable for NPPs

- Decree of the SÚJB No. 309/2005 Coll., on provision of technical safety for classified equipment
- Decree of the SÚJB No. 146/1997 Coll. Specifying Activities Directly Affecting Nuclear Safety and Activities Especially Important from Radiation Protection Viewpoint, Requirements on Qualification and Professional Training, on Method to be Used for Verification of Special Professional Competency and for Issue Authorisations to Selected Personnel, and the Form of Documentation to be Approved for Licensing of Expert Training of Selected Personnel, amended in Decree of the SÚJB No. 315/2002 Coll.
- Decree of the SÚJB No. 193/2005 Coll., on list of theoretical and practical areas forming a content of education and of preparation required for performance of regulated activities within the scope of power of the State Office for Nuclear Safety
Specific situation in the Czech Republic

- Existing legislation is not fully up-to-date for new NPPs
- Due to composition of the parliament in its last tenure there was no political will to make any “nuclear” decision
- Modifications of existing regulations are under preparation, mainly due to the need of harmonization with European legislation and/or with WENRA reference levels
- Decision to develop some regulatory (not mandatory) guides as temporary solutions
- Since modifications of the existing legislation in line with international progress are expected in the near future, international safety requirements should be carefully considered
Basic trends in improvements of Czech regulations on nuclear safety, radiation safety and emergency preparedness

- Enhancement of security measures
- New terms introduced: PIE, BDBA, severe accidents as a part of design basis
- Improved definition of safety systems
- Broader consideration of PIE, including PIE at shutdown regimes, PIE initiated in fuel storage systems and in waste storage, etc
- Enhanced consideration of defence in depth, considering BDBA and severe accidents
- Definition of design basis, requiring regular updating
- Requirements on deterministic and probabilistic analysis and ways on performing the analysis
- Requirement to perform PSA Level 1 and 2
- Classification and qualification of SSCs
Basic trends in improvements of Czech regulations on nuclear safety, radiation safety and emergency preparedness

- Capability to manage heat removal in case of severe accidents
- Improved requirements on fire risk analysis and fire protection
- Specification of external hazards to be considered in the design
- Additional I&C and monitoring, including severe accidents
- Operator actions not needed earlier than in 30 minutes, unless specially justified
- Continuous monitoring of operability of systems, fail-safe design
- Enhanced requirements on main and emergency control rooms, habitability also in case of severe accidents
- Availability of emergency support centre
- Consideration of ageing and burn-up in reactivity control systems
- Strengthening of requirements on ageing management
- Prevention of high-pressure melt through
Basic trends in improvements of Czech regulations on nuclear safety, radiation safety and emergency preparedness

- Requirements on role of secondary circuit in maintaining safety functions
- Enhancement of reliability of emergency power supply (consideration of single failure)
- Capability of the containment and its systems to cope with severe accidents, including tightness
- Monitoring and control of containment leakages, including severe accidents
- Strengthened capability of containment isolation systems
- Systems for management of flammable gases
- Prevention of the containment melting through
- Strengthened requirements on manipulation with spent fuel, including damaged fuel
- Requirements on analysis of DBA, BDBAs, PSA and fire risk on 4 attachments
Example of combination of various safety requirements into the Initial Safety Analysis Report

### Chapter 15.5 Safety analysis

<table>
<thead>
<tr>
<th>Chapter of SAR</th>
<th>195/99 Sb.</th>
<th>IAEA NS-R-1</th>
<th>IAEA GSR Part 4</th>
<th>WENRA 01/2008</th>
</tr>
</thead>
<tbody>
<tr>
<td>15.5 Deterministic analyses, normal operation, DBA, BDBA, severe accidents</td>
<td>(195/99, §4-(8) Kvalita a vhodnost výpočtových programů, používaných k analýzám, důležitým pro jadernou bezpečnost, musí být ověřena. (195/99, Příloha č. 1. F.) Při bezpečnostních analýzách událostí abnormálního provozu a projektových nehod se musí vycházet z toho, že ke zvládnutí analyzované projektové události, tj. k převedení reaktoru do stabilizovaného stavu, mohou být použity pouze bezpečnostní systémy, zařazené do bezpečnostních tříd v souladu s požadavky zvláštního předpisu) a se zaručenou spolehlivostí. Funkce jiných aktivních systémů se při zvládání událostí abnormálního provozu a projektové nehody mohou a musí uvažit pouze tehdy, jestliže zhorší průběh události. To znamená, že v průběhu zvládání událostí abnormálního provozu a projektové nehody se funkcí aktivních systémů, neklasifikovaných jako vybraná zařízení v souladu s požadavky zvláštního předpisu) neuváží, nebo se uvažuje jejich působení před vznikem a v průběhu událostí způsobem, který je pro zvládnutí události nejméně přínosný. (195/99, Příloha č. 1. G.) Při bezpečnostních analýzách událostí abnormálního provozu a projektových nehod se musí dále předpokládat: 1. Zapuštění bezpečnostních systémů na takové výkonové úrovni, která je pro průběh iniciační události nejméně přínosná......</td>
<td>(NS-R-1, 5.5.) Conservative design measures shall be applied and sound engineering practices shall be adhered to in the design bases for normal operation, anticipated operational occurrences and design basis accidents so as to provide a high degree of assurance that no significant damage will occur to the reactor core and that radiation doses will remain within prescribed limits and will be ALARA. (NS-R-1, 5.6.) In addition to the design basis, the performance of the plant in specified accidents beyond the design basis, including selected severe accidents, shall also be addressed in the design. The assumptions and methods used for these evaluations may be on a best estimate basis. (NS-R-1, 5.70) The computer programs, analytical methods and plant models used in the safety analysis shall be verified and validated, and adequate consideration shall be given to uncertainties. (NS-R-1, 5.71.) The deterministic safety analysis shall include the following: (1) confirmation that operational limits and conditions are in compliance with the assumptions and intent of the design for normal operation of the plant; (2) characterization of the PIEs (see Appendix I) that are appropriate for the design and site of the plant;......</td>
<td>(GSR 4, 4.48.) It has to be determined in the safety assessment whether there are adequate safety margins in the design and operation of the facility, or in the conduct of the activity in normal operation and in anticipated operational occurrences or accident conditions, such that there is a wide margin to failure of any structures, systems and components for any of the anticipated operational occurrences or any possible accident conditions. Safety margins are typically specified in codes and standards as well as by the regulatory body. It has to be determined in the safety assessment whether acceptance criteria for each aspect of the safety analysis are such that an adequate safety margin is ensured. (GSR 4, 4.54.) The aim of the deterministic approach is to specify and apply a set of conservative deterministic rules and requirements for the design and operation of facilities or for the planning and conduct of activities. When these rules and requirements are met, they are expected to provide a high degree of confidence that the level of radiation risks to workers and members of the public arising from the facility or activity will be acceptably low. This conservative approach provides a way of compensating for uncertainties in the performance of equipment and the performance of personnel, by providing a large safety margin......</td>
<td>(WENRA 01/2008, 8.1) The initial and boundary conditions shall be specified with conservatism. (WENRA 01/2008, 8.2) The worst single failure shall be assumed in the analyses of design basis events. However, it is not necessary to assume the failure of a passive component, provided it is justified that a failure of that component is very unlikely and its function remains unaffected by the PIE. (WENRA 01/2008, 8.3) Only safety systems shall be credited to carry out a safety function. Non-safety systems shall be assumed to operate only if they aggravate the effect of the initiating event. (WENRA 01/2008, 8.4) A stuck control rod shall be considered as an additional aggravating failure in the analysis of design basis events. ......</td>
</tr>
</tbody>
</table>
EU legislation is also a part of CR legal system

This legislation includes:

- Regulations: directly and generally legally binding documents, applicable for all EU countries,
- Directives: all EU countries are obliged to implement the requirements into their national legislation in a given time frame; it is up to the countries to decide the way of implementation,
- Decisions: legally binding acts, however only for a given country, company or an individual (binding for the addressee only),
- Recommendations/Opinions: not binding documents, countries may decide voluntarily whether or not to implement into their legislation,
- Soft law – informative documents such as “White Books”, “Green Books”.

Example of hierarchical structure of codes and standards

- **Level I**: National legislation
  - Legally binding
  - No deviations allowed

- **Level II**: IAEA Safety Fundamentals and Safety Requirements
  - IAEA Safety Guides
  - WENRA Reactor Safety Reference Levels
  - ISO Quality Assurance and Environmental Requirements
  - Deviations approved case by case before contract signature

- **Level III**: Nuclear Safety Regulations of the country of origin of the technology or an EU country, IAEA Safety Guides and ISO Guides

- **Level IV**: Nuclear Oriented Codes and Standards applicable in the country of origin of the technology, Nuclear Oriented Codes and Standards required by the Owner
  - Justified Deviations during the project development

- **Level V**: Conventional Codes and Standards proposed by the Supplier and Conventional Codes and Standards required by the Owner
  - Deviations are allowed with justification

Any consistent set of the internationally recognized codes/standards is acceptable (ANSI, ASME, RFS, BS, DIN, KTA, PNAE G, GOST)
Implications of large reactor power

- Power of reactors currently on the market 1150 – 1700 MWe, positive for plant economy
- No substantial impact on reliable heat removal: linear power is reduced in comparison with existing plants
- Issues associated with availability of cooling water: water intake for the plant cooling as well as process-generated waste water (blowdowns from the circulation cooling water circuit and the essential service water, waste water from neutralization plant) may be doubled
- Issues associated with the electrical grid: dynamic stability and need for back-up power; investments needed, additional high-voltage connections
Plant features associated with economy of operation

- Plant availability (91 – 95 %)
- Plant efficiency (gross up to 39 %, net up to 36 %)
- Fuel burn-up (60 – 70 MWd/kg)
- Number of unplanned reactor scram (<0.5 per year)
- Capability of on-line maintenance (redundancy of systems)
- Duration of refuelling/maintenance outage
- Number of operating personnel (220-350?)
- Collective radiation exposure (~0.5 manSv per unit/year)
- Volume and quality of solid radwaste produced (0.042-0.05 m3/MWe per year, acceptability for national depository)
- Transfer of know-how to local companies for future operational support
- Repair-ability of fuel assemblies
- Clear demonstration of plant lifetime, need and possibility for replacements of certain components during the lifetime
- Conditions for achieving good plant economy should be clearly specified
Operational flexibility

- Capability to operate at reduced power (e.g. 50-100 %)
- Capability of frequency control (e.g. ±3%, power range 50-100 %)
- Secondary and tertiary power control (e.g. ±10%, 1%/min, power range)
- Load follow operation (e.g. 20-30% up and down, 100 times per year, 0.5 %/min)
- Operability in an island mode, sufficient duration
- Possibility of using MOX fuel
- Possibility of stretch-out operation
- Possibility of using fuel from different suppliers
- Flexible fuel loading pattern
- Duration of fuel cycle (12 – 24 months)
- Using NPP for heat supply for towns nearby (up to 100 km)
- Capacity of the fresh fuel storage
- Capacity and location of on-site spent fuel storage
Compliance of plant features with local conditions

- Acceptability of standards of the country of origin vs national standards in sensitive areas, such as fire protection, radiation protection, safety of pressure vessels
- Adherence with the national quality assurance standards and rules
- Compatibility of the plant with national power grid and its grid code (operational limits, load follow capabilities, stability)
- Compliance of electrical circuits separation criteria, EMC and EMI issues with national standards
- Optimization of the circulating water system and cooling towers to site specific conditions
- Acceptability of radiological impact of releases, considering national legislation and presence of existing units on the site
- Possibility of sharing existing radwaste treatment technology with new needs, availability of sufficient storage capacity, acceptability of radwaste with disposal limits
- Compliance of the design with all site specific external hazards, including earthquakes, sabotage acts, etc
- Location of cooling towers and other facilities without conflict with other facilities on the site, in particular with electric power outlet.
- Utilization of existing facilities for construction of new units
Size of major plant components

- **AP 1000**: RPV mass 295.7 t, diameter 4470 mm, height 10256 mm, SG height 24.4 m, diameter 6096 mm, mass 664.1 t
- **EPR**: RPV mass 526 t, external diameter 5385 mm, height 12708 mm, SG mass 550 t, diameter 5168 mm, height 24621 mm
- **APWR**: RPV mass 590 t, external diameter 5800 mm, height 13600 mm, SG mass 420 t, diameter 5100 mm, height 21700 mm
- **AES-2006**: RPV mass 330 t, external diameter 4800 (without nozzles) mm, height 11185 mm, diameter 4490 mm, length 13820 mm

- Large components with the diameter ~4.5 – 6.1 m, length up to 24.6 m, and mass up to 664 t should be transported and adequate transport routes should be prepared
Proven design and novel design measures

- IAEA Safety Requirement, INSAG Basic Safety Principle: Wherever possible, structures, systems and components important to safety shall be of a design proven in previous equivalent applications.

- Novel components should be used only in case of clear advantages in specific areas (safety, cost-efficiency, performance, reliability): systems for severe accidents, passive systems, computer-based I&C.

- Acceptability of novel design solutions shall be demonstrated by tests and analysis, with results available for review.

- Both active and passive type design features are acceptable; for passive systems the issues of safety classification, application of the single failure criterion, reliability and testability of structures, systems and components, consideration of failures in safety analysis shall be addressed.

- To the extent possible, the concept of the standard design shall be followed; any deviation from the standard design should be known to the customer.
Anti-seismic plant resistance

- Available standard designs are made for horizontal acceleration 0.25-0.3g; this may be less than required for the given site and adequate reinforcement is necessary.
- As a general rule probability of occurrence of the safe shut-down earthquake (SSE) shall not be higher than 1E-4 by year.
- CDF should be less than 1E-5, therefore for an earthquake more severe than SSE it should be demonstrated that it does not lead to core damage.
- For earthquakes of higher intensity it shall be demonstrated that there will be no significant consequences (Seismic Margin Assessment).
Double containment

- New IAEA Safety Requirements for Design: Provisions has to be made for the collection and controlled release or storage of materials that may leak from the primary containment to the environment.

- EUR: The plant shall be equipped with a primary and a secondary containment.
  - The secondary containment is a fission product confinement envelope, which surrounds (entirely or only partially) primary containment.

- All containment functions can be fulfilled by a single containment, but secondary containment can significantly contribute to reduction of releases and protection against external hazards.

- Available designs: full scope double containment (EPR, AES-2006), partial secondary containment (AP-1000, APWR).

- Controlled venting of surrounding structures in emergencies should be ensured.
Containment by-pass

- Containment by-pass may represent significant contribution to large early releases and should be addressed in the design.
- By-pass of the secondary containment in case of severe accidents can represent very large contribution to external releases and doses (more than 90%).
- Probability of containment bypass sequences, with direct releases of primary coolant outside the primary containment, shall be minimized.
- By-pass of the secondary containment shall be quantified.
- In particular, primary coolant releases to the environment in connection with primary-to-secondary circuit leakages, including creep ruptures of steam generator tubes during severe accidents, shall be minimized.
Mitigation of severe accidents – ensuring containment integrity

- New IAEA Safety Requirements: the design has to include measures to avoid failure of the containment during severe accident sequences that can challenge the containment integrity.
- The best practice are the solutions with long-term stabilization of molten corium inside the containment and hydrogen management.
- Technical means should include measures for reliable fast depressurisation system to prevent core melting under high pressure.
- Corium stabilization: large flooded reactor cavity (APWR), in-vessel corium retention (AP 1000), cooled corium spreading compartment (EPR), core catcher (AES-2006).
- Hydrogen management: igniters (APWR, AP 1000), igniters and PARs (EPR), PARs (AES-2006).
- Attention shall be paid to demonstration of coolable configuration of molten corium, either inside or outside reactor vessel.
- Less stringent requirements on special equipment for mitigation of severe accidents are acceptable (qualification, single failure, seismic resistance, power supply, reliability).
Enhanced resistance against aircraft crash and against malevolent acts

- All international requirements (IAEA, EUR) suggest use of probabilistic and deterministic approach; if probability is low (<1E-7), design measures are not required
- Worldwide convergence of approaches not yet seen, methodology of analysis not fully established
- However, other circumstances in EU (political, regulatory) do not accept excluding fully a malevolent attack by terrorists
- Therefore, regardless low probability of random aircraft crash, protection against crash from small sporting aircraft up to large airplanes should be considered in the design deterministically
- Both primary (impact, induced vibrations) and secondary (fires, explosions) effects of an aircraft crash shall be considered
- Distinction can be made between design basis aircraft crash and beyond design basis aircraft crash, with different methodologies used in safety demonstration
- Protection against sabotage acts shall be considered in the design, with balanced use of safety and physical security measures
- Measures such as physical separation, redundancy, hardening, bunkering, secondary control room, etc. should be considered as part of the protection of the facility
Format and content of safety analysis report

- Two options are available
  - RG 1.70 Standard format safety analysis reports for nuclear power plants as updated with requirements of RG 1.206
- Only Slovak regulatory body has issued a guidance document, setting the format and content in accordance with IAEA Safety Guide GS-G-4.1
- PSAR for Mochovce 3&4 in Slovakia was successfully completed accordingly
- However, all major reactor vendors follow different derivatives of RG 1.70; change would be certainly possible for additional time and costs
- IAEA starts developing an updated Safety Guide
- The issue should be discussed with the national regulatory body
Demonstration of adequacy of safety margins

- IAEA Safety Standards and regulatory guides: Computer codes used for demonstration of safety should preferably be of best-estimate type, validated to the extent possible in the area of intended code use.

- Computer codes shall be at least briefly described in the SAR, including summary of code validation status, with a reference made to complete documentation of the code. On request, validation reports shall be made available for review.

- BE codes should be used in a safety analysis either in combination with a reasonably conservative selection of input data or in combination with realistic assumptions on input data associated with evaluation of uncertainties.

- Uncertainties that may have implications for the outcome of the safety analysis and decisions made on that basis shall be addressed in uncertainty and sensitivity analyses.

- In the safety documentation sufficient information shall be provided to allow an independent verification of safety margins.
Safety classification of equipment

- All SSCs important for safety shall be identified and classified on the basis of their importance for safety.
- Systems of classification are different in different vendor countries recipient countries.
- Cross-reference table should be developed to specify interrelation of various classification systems.
- IAEA: Interrelation between safety functions and safety related systems shall be done in a systematic way.
- The classification shall identify for each safety class at least the following:
  - The appropriate codes and standards in design, manufacturing, construction and inspection,
  - Need for emergency power supply,
  - Qualification to environmental conditions,
  - Seismic classification,
  - The availability or unavailability status of systems for PIEs to be considered in deterministic safety analysis,
  - The QA provisions.
Methodology for safety demonstration in case of severe accidents

- Severe accidents should be considered in the design, although demonstration of safety may be more relaxed
- There is consensus on acceptability of “best-estimate” analysis
- Significant differences between US and European approaches
- US approach: strongly probabilistic without specified acceptance criteria
- European approach: deterministic using a reference accident (s) and demonstrating compliance with the criteria
- Deterministic acceptance criteria used for beyond design basis accidents including severe accidents seems better acceptable for Europe
- Deterministic radiological criteria shall be expressed either in terms of releases or in terms of potential radiation doses to the public and the site personnel (doses calculated by approved national methodology)
- Methodology for calculation of radiological consequences shall be specified
Conclusions

- Development of a nuclear project from the first decision to the operation is a complex process, consisting of many steps.
- Duration of this process, even with infrastructure available, is more than 10 years (13 years is current estimate in the CR).
- A vendor shall typically be selected in an open tender.
- Development of a Feasibility Study and Bid Invitation Specification are important initial steps in selection of a vendor.
- Use of European Utility Requirements or another similar document is a good starting process, with modifications taking into account national needs and conditions.
- Ensuring nuclear safety is a necessary condition for future success, it should be therefore adequately covered in the documents from very beginning.
- Compliance with national nuclear safety regulations is a must, but other considerations should be made, too.
- Examples of such considerations were provided in the presentation.