Comparison between International and Slovak Design Safety Requirements on Severe Accidents and Feasibility Analysis of the Safety Enhancement of VVER 440/V213 Plants to Comply with New Safety Requirements

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Abstract. The paper analyses and presents the comparison of current international requirements on severe accidents (IAEA Safety Standards, WENRA Reference Levels, EUR) and relevant Slovak national legislation. The comparison shows that there is good consistency between WENRA Reference Levels, IAEA Safety Requirements, and Slovak Legislation. It is found that the WENRA Reference Levels are reasonably-balanced requirements, applicable for both existing and new designs, as compliance with WENRA also ensures general compliance with IAEA Safety Requirements for design as well as with the Slovak Legislation. Furthermore, the paper presents the possibility of safety upgrade for VVER 440/V213 units in order to comply with WENRA Reference Levels related to severe accidents.

1. INTRODUCTION

In accordance with the defence-in-depth philosophy, severe accidents have to be considered in the nuclear power plant (NPP) design independently of any measures implemented for their prevention. Severe accident mitigation features were not included in the original design of VVER 440/V213 NPPs. This paper analyses and presents the comparison of current international requirements on severe accidents and the Slovak national legislation to demonstrate that for VVER 440/V213 NPPs it is feasible to implement design upgrading measures which ensure compliance with the new safety requirements. Relevant measures for corium stabilization, hydrogen management, source-term reduction, containment overpressure protection, and long-term heat removal are discussed in the paper with reference to Mochovce NPP Units 3&4 currently under construction in Slovakia.

2. COMPARISON OF NATIONAL AND INTERNATIONAL SAFETY REQUIREMENTS

The Slovak Republic, by ratifying the Convention on Nuclear Safety, has committed itself to continuous safety improvements of NPPs. In the Slovak Atomic Act No 541/2004 and relevant Decrees the utility is obliged “to perform regular, comprehensive and systematic assessments of nuclear safety, taking into account the state of the art in the area of nuclear safety review, and to take measures to eliminate any deficiencies identified”. More specifically, legal requirements referring to severe accidents in design and operation of NPPs are set up in the Slovak Decree No. 50/2006 and other associated Decrees on Laying Down Details of the Requirements for Nuclear Safety of Nuclear Installations. The requirements specified in the above mentioned Decree include hardware measures and development of relevant procedures. This trend is in accordance with the revised set of the International Atomic Energy Agency (IAEA) Safety Standards; in particular, safety requirements for NPP design [1], safety guides for the design of reactor containment systems [2] and severe accident management programmes [3].

Special attention is also paid to Western European Nuclear Regulators’ Association (WENRA) Reference Levels [4], developed on a consensual basis of the involved Regulatory Bodies and meant to be the common basis of safety requirements for existing NPPs. According to 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. Although current WENRA Reference Levels represent only recommendations for safety of existing plants, it is advisable to follow them, since they are expected to become Regulatory requirements to be implemented in the near future.

Currently, new nuclear plants are also being designed and constructed with increased performance and a higher safety level, in accordance with utilities’ expectations, such as expressed in European Utility Requirements (EUR) [5] (which, however, are mainly to be considered for standardized designs to be licensed in different countries). Owing to the general practice in the nuclear industry related to the implementation, as far as reasonable of new safety requirements into the design of operating plants, it is nevertheless advisable to consult the currently-valid documentation of existing plants as well. In particular, this is important due to another general trend, i.e. to ensure long-term operation of existing plants significantly beyond the originally-envisaged lifetime of the plants.

National and international safety requirements in Slovakia can hierarchical be set as summarized below:
1. Slovak legislation and relevant international treaties and conventions,
2. IAEA Safety Requirements,
3. WENRA Reference Levels,
4. IAEA Safety Guides.

Detailed comparison of the requirements listed above in the area of severe accident mitigation measures was performed. For the comparison the following areas were considered:
- General requirements,
- Selection of beyond design basis accidents (BDBA)/severe accidents,
- Analysis of BDBA/severe accidents,
- Monitoring of plant parameters,
- Presentation of plant parameters in the main control room (MCR),
- MCR habitability,
- Equipment survivability,
- Containment isolation,
- Containment leakages and containment by-pass,
- Prevention of high-pressure core melt scenarios,
- Management of combustible gases,
- Prevention of containment degradation by molten fuel,
- Long-term containment heat removal,
- Containment protection against long-term over-pressurization,
- Source term reduction,
- Severe accident management guidelines (SAMG) development and implementation,
- Elimination of highly energetic events.

The Table 1 below shows for some areas an example of the comprehensive comparison performed. In this table comparison of WENRA Reference Levels, IAEA Safety Requirements on Design, and the Slovak Decree No. 50/2006, considered as obligatory rules for design enhancements, is shown. Similar comparison table was made for IAEA Safety Guides [2, 3] and for EUR. Owing to the very large volume of the complete comparison tables, they can not be presented in the paper.

From the comparison it was concluded (as can be partially seen from table 1), that there is good consistency between WENRA Reference Levels, IAEA Safety Requirements, and Slovak legislative requirements. In particular, the statements in IAEA Safety Requirements and in the Slovak Decree No. 56/2006 and associated other Decrees are very similar. However, wording of WENRA is more stringent using statements such as “...shall exist”, “...shall be possible”, etc. rather than “adequate consideration shall be given…” used by IAEA. WENRA strictly requires that “high pressure core melt scenarios shall be prevented”, while in IAEA documents an equivalent “should” statement is included
in lower level Safety Guide on design of the containment. The most questionable WENRA requirement is “Containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable”, which allows ambiguous interpretations.

Similar wording was found in IAEA Safety Guide (on Containment) and in EUR. No contradictions were identified, but there are differences in terminology used and in level of details. More details are included in the EUR, such as:

- Quantitative specification of deterministic and probabilistic targets (not specified in IAEA Standards)
- Specification of certain computational methods (hydrogen, containment loading, radiation doses, etc)
- Prescription of some engineering solutions to address the challenges.
<table>
<thead>
<tr>
<th>Issue</th>
<th>WENRA Reference Levels</th>
<th>IAEA Safety Requirements on Design</th>
<th>Slovak Decree No. 50/2006 Coll.</th>
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<tbody>
<tr>
<td><strong>Presentation of plant parameters in the MCR</strong></td>
<td>Necessary information from instruments shall be relayed to the control room as well as to a separately located supplementary control room/post and be presented in such a way to enable a timely assessment of the plant status and critical safety functions in severe accident conditions.</td>
<td>In the systems manager role, the operator shall be provided with information that permits the following: (1) the ready assessment of the general state of the plant in whichever condition it is, whether in normal operation, in an anticipated operational occurrence or in an accident condition, and confirmation that the designed automatic safety actions are being carried out;</td>
<td>Nuclear installations shall be equipped with operational control rooms, where the nuclear installation can be safely and reliably monitored and controlled in all states.</td>
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<td><strong>MCR habitability</strong></td>
<td>A control room shall be provided from which the plant can be safely operated and from which measures can be taken to maintain the plant in a safe state or to bring it back into such a state after the onset of anticipated operational occurrences, design basis accidents and severe accidents. Appropriate measures shall be taken and adequate information provided to safeguard the occupants of the control room against consequent hazards, such as undue radiation levels resulting from an accident condition or the release of radioactive material or explosive or toxic gases, which could hinder necessary actions by the operator. Appropriate measures shall be taken to protect the occupants (of emergency center) for a protracted time against hazards resulting from a severe accident.</td>
<td>Control rooms shall be designed so that, with respect to protection of the health of employees at work, they can be accessed and occupied in a safe and healthy manner even under accident conditions. The design shall include ergonomic principles including man-machine interfaces.</td>
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<td><strong>Equipment survivability</strong></td>
<td>To the extent possible, equipment (such as certain instrumentation) that must operate in a severe accident should be shown, with reasonable confidence, to be capable of achieving the design intent.</td>
<td>Installations inside containment systems shall be designed so as to fulfil their functions and so that their impact on other systems, structures and components is limited.</td>
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<td><strong>Containment isolation</strong></td>
<td>Isolation of the containment shall be possible in a beyond design basis accident.</td>
<td>Adequate consideration shall be given to the capability of isolation devices to maintain their function in the event of a severe accident. Adequate consideration shall be given to the capability of containment air locks to maintain their function.</td>
<td>Furthermore, account shall be taken of the ability to moderate the consequences of selected severe accidents and limit the escape of radioactive substances into the natural environment.</td>
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<tr>
<td>Topic</td>
<td>Description</td>
<td>Action</td>
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<td>Containment leakages and containment by-pass</td>
<td>The leak-tightness of the containment shall not degrade significantly for a reasonable time after a severe accident. However, if an event leads to bypass of the containment, consequences shall be mitigated.</td>
<td>Adequate consideration shall be given to the capability to control any leakage of radioactive materials from the containment in the event of a severe accident. Adequate consideration shall be given to the capability of penetrations to remain functional in the event of a severe accident.</td>
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<td>Management of combustible gases</td>
<td>Combustible gases shall be managed in a severe accident. In particular, the effects of any predicted combustion of flammable gases shall be taken into account. Adequate consideration shall be given to the control of fission products, hydrogen and other substances that may be generated or released in the event of a severe accident.</td>
<td>The containment shall be fitted with systems to monitor hydrogen and radioactive substances, which might be released into it during postulated initiating events and after the occurrence of them. Together with other systems, these systems shall a) reduce the volumetric activity and regulate the composition of fission products, b) monitor and maintain the volume concentration of hydrogen at the permissible values so as to ensure the integrity of the containment</td>
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<td>Prevention of containment degradation by molten fuel</td>
<td>Containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable. Provision for maintaining the integrity of the containment in the event of a severe accident shall be considered. Adequate consideration shall be given to the capability of internal structures to withstand the effects of a severe accident. Adequate consideration shall be given to extending the capability to transfer residual heat from the core to an ultimate heat sink so as to ensure that, in the event of a severe accident, acceptable temperatures can be maintained in structures, systems and components important to the safety function of confinement of radioactive materials.</td>
<td>Nuclear installations shall be equipped with containment systems which, during postulated initiating events associated with the escape of radioactive substances and ionising radiation into the environment, limit such releases so that they are less than the set limit values for releases, unless this function is performed by other facilities.</td>
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<td>Long-term containment heat removal</td>
<td>Pressure and temperature in the containment shall be managed in a severe accident. Adequate consideration shall be given to the capability to remove heat from the core following a severe accident. Adequate consideration shall be given to the capability to</td>
<td>Installations involved in the removal of heat released by fission and residual heat shall be designed so as to provide for reliable cooling of materials in all states. Heat removal systems</td>
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remove heat from the reactor containment in the event of a severe accident.

shall be backed up, physically separated and isolated and shall be able to be interlinked so that they can perform their functions during a single failure. The design shall include a solution for reliable final heat removal from selected installations during states of normal operation, abnormal operation, design-basis accidents and, during selected severe accidents make a partial contribution to heat removal. Final heat removal shall mean the removal of residual heat into the atmosphere or into water or a combination of them.
3. OVERVIEW OF IMPLEMENTATION OF SEVERE ACCIDENT MITIGATION MEASURES IN EXISTING REACTORS ABROAD

Based on comparison of status in implementation of severe accident mitigation measures in different countries (with reference to pressurized water reactor technology and in particular on VVER 440 reactors) it is possible to state the following:

- Independently of whether or not it is formally required by the Legislation, the majority of utilities have decided to address severe accidents in operation of existing plants,
- Nearly all plants recognized the importance of reactor coolant system fast depressurization, performed either by existing plant features (pressurizer relief and safety valves) or, more rarely but more reliably, by dedicated depressurization means,
- Use of passive autocatalytic recombiners is a prevailing way for hydrogen management; in some cases by relying on hydrogen ignition capability of the recombiners and more rarely in combination with special hydrogen igniters (Loviisa),
- Reactor cavity flooding is considered as a helpful strategy, independently whether or not there is confidence in in-vessel retention capability or ex-vessel corium coolability,
- In-vessel retention strategy, if feasible, seems to be considered as an efficient way for halting accident progression and reduction of consequences; this strategy is applied in VVER 440 reactors,
- Several countries (Germany, France, Sweden) implemented filtered containment venting independently whether relying or not on corium coolability inside the containment,
- VVER 440 NPPs have plans to implement severe accident mitigation measures, although the status of implementation varies significantly. Loviisa has performed a comprehensive implementation of all measures, and shall be considered as an example of good practice.

4. DESIGN TARGETS (UTILITY ACCEPTANCE CRITERIA) FOR SEVERE ACCIDENTS

There are no quantitative acceptance criteria in the Slovak Legislation especially addressing severe accidents at NPPs. Therefore, in the case of Mochovce 3&4 design, certain quantitative requirements in terms of design targets rather than acceptance criteria were specified. The targets for radiological impact to the environment, public and workers were given in the form of dose limits. The general objective of the targets is to minimise the requirements for emergency planning and off-site countermeasures, but some lighter actions like iodine prophylaxis and sheltering are considered acceptable for severe accidents.

There were three objectives specified related to radiological impact on the public:

- No emergency protection actions beyond exclusion zone, i.e. actions involving public evacuation, based on projected doses up to 7 days, which may be implemented during an emergency phase of an accident, e.g. during the period in which significant releases may occur.
- No delayed actions beyond exclusion zone, i.e. actions involving public temporary relocation, based on projected doses up to 30 days caused by ground-shine and aerosol re-suspension, which may be implemented after the practical end of radioactivity release phase of an accident.
- No long term actions beyond exclusion zone, i.e. actions involving public permanent resettlement, based on projected doses up to 50 years caused by ground-shine and aerosol re-suspension. Doses due to ingestion are not considered in this definition.

In addition, in compliance with EUR it was required to address the long-term effects of the accident by limiting releases of Cs 137 (which is the most significant radioisotope for long-term effects) to 100 TBq.

For ensuring containment integrity as the last barrier against releases of radioactivity the following derived design targets were established for Mochovce 3&4 design:

- At the time of core relocation the pressure inside the reactor coolant system shall be below 2 MPa in order to avoid high pressure melt ejection.
- Reactor pressure vessel integrity shall be ensured by means of external cooling of the vessel (in-vessel retention strategy).

5. ESSENTIAL HARDWARE MEASURES TO BE IMPLEMENTED IN VVER 440/V213 DESIGN

Scope of hardware measures implemented in the design of Mochovce 3&4 units resulted from broad range considerations, including:
- Full compliance with current national legislation and most relevant international requirements (WENRA Reference Levels and IAEA Safety Requirements),
- Compliance to the possible extent with international recommendations (IAEA Safety Guides) and EUR,
- Deterministic safety analyses (thermal-hydraulic, structural, radiological), feasibility of implementation, taking also into account the results of PSA studies,
- Feasibility of implementation in the partially built NPP.

The proposal of the measures is addressing severe accident mitigation generally, without focusing to a specific, selected severe accident scenario. For this purpose particularly challenging bounding scenarios have been identified for the evaluation of the effectiveness of the proposed measures and modifications. As the representative of fast-evolving scenarios with maximum decay power in the severe accident phase, the loss of coolant accident (LOCA) with large break diameter was identified. In addition to the LOCA initiating event, several serious malfunctions impairing core cooling were assumed to reach maximal-challenging conditions, generally up to the long-term blackout of the unit. Second representative bounding scenario was the total loss of power supply (black out, loss of both off site and on site power). This scenario is representative for slowly progressing severe accidents, often associated with high pressure in the reactor coolant system (RCS).

The overview of hardware measures implemented in the new Mochovce 3&4 design is shown in table 2. Safety upgrading is based on retention of molten corium in the reactor pressure vessel (RPV) ensured by flooding of the reactor cavity (see Fig. 1), of course complemented by many other measures aimed at maintaining containment integrity.

Table 2. Hardware measures dedicated to mitigation of severe accidents in Mochovce 3&4 VVER440/V213

<table>
<thead>
<tr>
<th>Issue</th>
<th>Measure</th>
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<tbody>
<tr>
<td>Prevention of HPME</td>
<td>Controlled depressurization of RCS during severe accident by means of an additional branch with two closing valves</td>
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<tr>
<td>Hydrogen management</td>
<td>Qualified monitoring of hydrogen concentration and installation of recombiners and igniters</td>
</tr>
<tr>
<td>Prevention of sub-atmospheric pressure</td>
<td>Vacuum breakers installed between air traps and steam generator (SG) compartments</td>
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<td>In vessel retention of the corium</td>
<td>Provision of sufficient coolant inventory for cavity flooding by means of draining of the bubble condenser trays</td>
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<td>Modification of thermal shielding of the bottom of the reactor pressure vessel in order to enhanced external cooling of the RPV</td>
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<td>Establishing circulation channel for coolant along the reactor pressure vessel wall between the SG compartments and the reactor cavity</td>
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<td></td>
<td>Modification of the drain line from the reactor cavity to ensure tightness of the reactor cavity</td>
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</tbody>
</table>
Management of open reactor sequences | Delivery pump and line from the dedicated external tanks for supply of coolant into the low pressure emergency core cooling system (ECCS) and the spent fuel pool
---|---
Reduction of containment source term | External sources of coolant for containment spray and radioactivity wash-down in the early phase of a severe accident, consisting of tanks, pump and associated piping system
Long-term heat removal | Use of existing spray system for long term containment heat removal
Reliable power supply | Dedicated diesel-generator for severe accident management

| I&C for severe accidents | Dedicated I&C severe accident control panel |
| Control room habitability | Dedicated ventilation system of the main control room |
| Availability of parameters for control of severe accidents | Monitoring of parameters needed for accident management |

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**Fig. 1. Overall arrangement for Mochovce 3&4 reactor cavity flooding and containment spraying**

For existing operating VVER440/V213, the implementation of severe accident mitigation features has to be developed in subsequent phases during outages. Single features have to be analysed by taking into account their possibility to be actually implemented based on resulting complexity, realistic timing for preparation of investments, existing rules for procurement and investments (as investment cost have also be taken into account). Therefore sequences of implementation have to be defined by competent Design Organizations which can effectively identify all interfacing consequences after receiving the boundary limits evidenced by the operation personnel and plant owner managers.
6. COMPLIANCE OF THE PROPOSED MEASURES WITH WENRA REFERENCE LEVELS

Cross-reference of individual proposed measures to relevant WENRA Reference Levels (10 relevant requirements identified) is as follows:

6.1. Adequate instrumentation shall exist which can be used in severe accident environmental conditions in order to manage such accidents according to guidelines/procedures for severe accidents.

Relevant measures:
- Monitoring of coolant level in reactor cavity,
- Monitoring of coolant level in steam generator boxes,
- Modification of pressure monitoring in steam generator boxes,
- Monitoring of containment temperature,
- Monitoring of hydrogen concentration inside containment,
- Pressure monitoring inside air traps,
- Temperature monitoring inside air traps,
- Core outlet temperature measurement,
- Pressure measurement inside reactor pressure vessel,
- Monitoring of primary-secondary pressure difference,
- Monitoring of radioactivity inside containment,
- Diverse additional monitoring chains.

6.2. Necessary information from instruments shall be relayed to the control room as well as to a separately located supplementary control room/post and be presented in such a way to enable a timely assessment of the plant status and critical safety functions in severe accident conditions.

Relevant measures:
- Dedicated I&C severe accident control panel
- Dedicated ventilation system of the main control room.

6.3. Isolation of the containment shall be possible in a beyond design basis accident.

Relevant measures (in addition to measures under Reference Levels No. 4, 7, 8, 9):
- Modification of penetrations leading into the reactor cavity,
- Modification of the access cavity door,
- Modification of the drain line from the reactor cavity,
- Dedicated I&C severe accident control panel.

6.4. The leaktightness of the containment shall not degrade significantly for a reasonable time after a severe accident. However, if an event leads to bypass of the containment, consequences shall be mitigated.

Relevant measures (in addition to measures for pressure and temperature management in containment):
- Vacuum breaker,
- Delivery pump and line for supply of coolant into the spent fuel pool,
- VVER 440/213 reactors are rather insensitive to the induced steam generator tube rupture due to configuration of the hot legs with loop seals,
- Containment bypass induced by a severe accident is also prevented by controlled depressurization of the reactor coolant system.
6.5. **High pressure core melt scenarios shall be prevented.**

Relevant measures:
- Controlled depressurization of the reactor coolant system during a severe accident.

6.6. **Combustible gases shall be managed in a severe accident.**

Relevant measures:
- Installation of recombiners and igniters.

6.7. **Containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable.**

Relevant measures:
- Modification of penetrations leading into the reactor cavity,
- Modification of the access cavity door,
- Modification of the drain line from the reactor cavity,
- Provision of sufficient coolant inventory for cavity flooding (draining system of the bubble tower trays),
- Dedicated diesel-generator,
- External sources of coolant,
- Establishing circulation channel for coolant along the reactor pressure vessel wall,
- Modification of thermal shielding of the bottom of the reactor pressure vessel.

6.8. **Pressure and temperature in the containment shall be managed in a severe accident.**

Relevant measures:
- Use of existing spray system in combination with external sources of coolant,
- Installation of recombiners and igniters,
- Dedicated diesel-generator.

6.9. **The containment shall be protected from overpressure in a severe accident.**

Relevant measures (in addition to measures for pressure and temperature management):
- Use of existing spray system,
- Use of existing ventilation system.

6.10. **The design extension analysis shall examine the performance of the plant in specified accidents beyond the design basis, including selected severe accidents, in order to minimise as far as reasonably practicable radioactive releases harmful to the public and the environment in cases of events with very low probability of occurrence.**

Relevant measures (in addition to measures for pressure and temperature management, containment isolation and containment leaktightness):
- External sources of coolant,
- Use of existing spray system,
- Dedicated diesel-generator.

As can be partially seen from the comparison above, implementation of the proposed hardware measures will allow reaching full compliance with the WENRA Reference Levels. In accordance with the results of comparison of various national and international safety requirements, this also means compliance with the Slovak Atomic Act and associated decrees, with IAEA Safety Requirements for the design and for operation.
7. CONCLUSIONS

From the comparison of various international safety requirements in the area of severe accident mitigation it can be concluded that the WENRA Reference Levels represent reasonably balanced requirements applicable for both existing and new designs. Compliance with WENRA also ensures full compliance with the IAEA Safety Requirements for design and for operation of NPPs as well as with the current Slovak legislative documents. The compliance is also true for relevant IAEA Safety Guides in the scope required for existing reactors.

Severe accident mitigation features were not included in the original design of VVER 440/V213 nuclear power plants, but the current trend is to include such features in the design, in order to make such power plants compliant with the most recent safety requirements. New national and international safety requirements in the area of severe accident mitigation have been comprehensively implemented into new design of Mochovce NPP, Units 3&4 equipped with VVER440/V213 reactors. In the process of development of the revised design, a significant level of consensus was reached among the stakeholders including recognized international experts regarding the reasonable and feasible scope of measures. The selected measures cover: corium stabilization in the RPV, hydrogen management, source-term reduction, containment overpressure protection, and long-term heat removal. The implementation of the selected measures is feasible with present-day engineering practices. Implementation of hardware measures will also allow implementing the effective severe accident management guidelines.

In addition, most of the hardware measures implemented in the Mochovce Units 3&4 appear feasible also for existing VVER 440/V213 units. Although comprehensive design considerations are still ongoing, it was demonstrated that implementation of the required measures in existing reactors is feasible and allows reducing radiological risk from severe accidents close to the level required for current evolutionary reactors.

REFERENCES