Proposal of In-vessel Corium Retention Concept for Paks NPP

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1. Introduction

The in-vessel corium retention (IVR) via external reactor vessel cooling (ERVC) seems to be a promising severe accident management (SAM) strategy not only for new generation of advanced PWRs, but also for VVER-440/V213 reactors, which were designed several years ago. In the case of advanced PWRs, significant attention was paid at the design stage to optimisation of ERVC loop in order to ensure efficient cooling of external reactor vessel surface in natural circulation regime. Contrary to the above-mentioned advanced designs, the VVER-440/V213 reactors were designed and built according to earlier standards and the possibility of core melting was not accounted for in the design stage of these low-powered reactors. Thus, the possibility of additional implementation of technical modifications, which are necessary for the creation of an ERVC loop, is rather limited. However, this shortcoming is compensated with significantly lower maximum heat fluxes than expected for higher-powered reactors.

2. Decay heat removal during IVR

The most limiting condition regarding the thermal loading occurs when the corium in the lower head is fully molten (except of the crust on the boundary of oxidic layer) and no decay heat is consumed in melting or heating up the debris or the vessel internals. Such a bounding case may not occur in a severe accident, but was chosen from licensing considerations. The melt or the RPV wall has to be cooled either internally or externally in order to prevent RPV failure. The operator would flood the vessel containing molten pool in lower head if any water injection is available. But the crust formed would drastically limit the transfer to conduction limit, and the internal cooling may not be effective enough.

External cooling is a more reliable way to keep outside wall temperature cool and thus to prevent its failure and retain the melt inside. Global efficiency of this heat transfer mechanism is higher than the upward heat transfer to coolant above the melt pool. Contrary to internal cooling in this case there is directly cooled the RPV wall, i.e. the wall which integrity should be preserved. Furthermore, this mechanism do not rely on recovery of water supply into RPV. When properly designed, the external cooling circuit is capable to remove the decay heat in passive natural circulation regime conserving the coolant inventory in the system. The success of IVR depends on the thermal margins: that means the difference between the thermal loading (by the melt natural convections) on the inside and the limits of coolability (due to boiling crisis) on the outside of RPV.

3. Upgrading of Paks NPP for severe accidents

The Hungarian Paks NPP, equipped with four VVER-440/V213 reactors, is in operation since 1982. The target of the accident management program is to increase the overall capability of the plant to
respond and recover from a severe accident situation. This capability could be increased by hardware modifications and with a guide to use the available resources in an optimal way. The key element of this program is SAMG (Westinghouse type) that are already under development. The level 2 PSA has been used to assess the risk impact and the risk reduction efficiency of selected SAM actions and to identify reasonable design basis for mitigative systems.

The SAM program approved by the regulator contains a set of planned plant modifications. The most important ones are described as follows.

*Primary circuit depressurisation* is necessary measure in order to ensure low-pressure core meltdown sequences and to reduce the risk for induced steam generator tube rupture through the circulation of hot gases. In case the in-vessel retention of corium strategy is selected, a low primary system pressure is also a definite requirement. The recent depressurization capability would reduce the pressure sufficiently and it was designed for potential releases of steam, two-phase mixture and water. In order to ensure sufficient opening reliability an independent so-called SAM power supply of the valves has to be installed.

The PSA results indicate that the risk of large releases dominated by *containment bypass* sequences that caused leaks from primary to secondary side of the steam generators. It is an effective precaution against containment bypass to implement blow-down lines on the bottom of the steam generators that are directed to the containment.

*Severe accident hydrogen* is confirmed as a major threat to containment integrity. The rapid onset of flammable conditions in an unmitigated severe accident necessitates a means of control. With the help of level 2 PSA it was showed that implementation of about 30 large passive autocatalytic recombiners would ensure that the containment would not experience high pressures loads in all those sequences that dominate the overall risk.

Both, *in-vessel corium retention* or ex-vessel debris cooling can only be accomplished by the active cavity flooding. It is already clear that IVR is potentially feasible but the potential for coolability of corium or core debris on the concrete base mat is still under investigation. There are double hermetic steel doors with rubber sealing in the sidewall of reactor cavity, which is a part of hermetic boundary. In case of ex-vessel cooling the thermal protection of those doors against temperature loads would have been solved also. In order to avoid a number of specific loading mechanisms caused by the eventual melt ejection into the reactor cavity the necessary plant modifications are needed for corium localisation and stabilisation inside reactor vessel gets higher priority in the SAM programme.

*Filtered venting* is used to prevent late containment failure. It is an effective precaution against late containment overpressure to modify operating confinement vent system to use as filtered venting.

The recent paper describes in more detail IVR concept and the plant modifications planned to implement.

4. **In-vessel retention for VVER-440/V213 NPPs**

The basic idea of in-vessel retention of corium is to prevent RPV failure by flooding the reactor cavity so that the reactor pressure vessel is submerged in water up to its support structures, and thus the decay heat can be transferred from the corium pool through the vessel wall and into the water surrounding the vessel and hence into confinement.
Necessary condition for keeping efficient heat transfer on the external surface of RPV is that the boiling crisis avoided and the heat transfer mode remains in subcooled single-phase convection or nucleate boiling mode. Film boiling is characterised by considerably lower heat transfer coefficient. Thus, if the heat transfer mode permanently changes to film boiling, then the surface temperature of the wall increases dramatically, which would sooner or later lead to failure of the RPV. However, the large thermal inertia of RPV wall stabilises the wall temperature and prevents fast temperature escalations of the surface even in the case of temporary local drying of the surface. Due to three-dimensional heat conduction in thick-walled RPV the area on which the film boiling occurs has to be of a substantial size.

The thermal loading on the inside is basically given by corium mass, corium composition, thermal power and shape of the lower head. Other factors, such as history of core melting, corium oxidation, radiative heat transfer from upper metallic layer, etc. may influence the thermal loading, too. In a typical estimation of thermal loading it is assumed that the whole fission power except of volatiles is relocated into lower head. This is clearly pessimistic assumption, typical for licensing kind of analyses.

For an operating reactor there are only limited possibilities to design the cooling loop. Such loop should enable either single or two-phase flow circulation in natural circulation regime and should be mass-conserving (i.e. assure return of condensed coolant from confinement into reactor cavity via “downcomer” of ERVC loop). It is obvious that the cooling efficiency of such additionally implemented ERVC loop will be lower than the efficiency of the loop optimised in the design stage of advanced PWR designs.

In particular case of VVER-440/V213 reactors the most important design features, favourable for adoption of the IVR concept, are as follows:

- low thermal power;
- large coolant volumes in both, primary and secondary systems (long elapsed time before the onset of core melting);
- reactor pressure vessel without penetration in lower head;
- massive stainless steel vessel internals, including massive core support plate (obstacle for early corium attack to RPV wall, additional heat sink due to capacity and latent heat, “dilution” of decay heat, thick metallic layer expected on the top of the molten pool);
- large water volume in lower head (long time since the start of core relocation to molten pool formation in lower head);
- high RPV and, consequently, high driving head for natural circulation in ERVC loop;
- presence of large diameter ventilation ducts of cavity cooling system (Fig. 1.) which may be used – after certain modifications – for cavity flooding and return of coolant from the confinement floor (downcomer of ERCV loop);
- annulus gap (width ~30 cm) between RPV and thermal insulation of cylindrical part of RPV (riser of ERCV loop).
Fig. 1. Two proposals of IVR for VVER – 440/V 213, variant with lowering of thermal/ biological shield (left) and variant with central hole in thermal shield (right)
On the other hand, negative features for adoption of the IVR concept are two main obstacles for creation of efficient ERVC loop:

- presence of massive thermal/biological shield (Figs. 1. and 3.) of lower head which prevent coolant access to RPV wall (the gap width between the RPV and this shield is only ~2 cm);
- flow restriction for steam-water mixture venting at the outlet from reactor cavity to confinement (gap ~3 cm due to presence of RPV support structures).

The shield can be in principle removed, as in Loviisa NPP or modified, as planned for Slovakian and Hungarian VVER-440 units (Fig. 1.). These two solutions and some theoretical assumptions have been already analyzed in the paper. In this paper previous activities performed for Paks NPP in 2006 of VUEZ and IVS Company (Slovakia) on the area of IVR conception are also presented.

5. **IVR proposal for Paks NPP**

Presence of coolant pool on the confinement floor is assumed during most of accidents. However, relatively large amount of coolant (~750 m$^3$) is necessary to reach the elevation of reactor cavity flooding valves (+ 6.80 m) and to flood the reactor cavity (Fig. 2.). The inherent water spilling from the bubbler condenser barbotage trays is achieved only for larger LOCAs. Manual draining (opening of drainage valves by operator) is required for smaller break diameters in order to gain missing coolant for cavity flooding. Due to limited diameter of drainage lines this action takes about 70-80 minutes.

![Flooding curve of SG box and connected system](image)

**Fig. 2. Flooding curve of SG box and connected system**

Water from the confinement floor enters through two new active inlet valves into large-diameter ventilation ducts, which are discharging into lower part of reactor cavity (Fig. 1.).

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Provisions (isolation siphon) should be also installed to avoid coolant flow through ventilation ducts into ventilation centre.

Modification of the thermal/biological shield of RPV lower head is necessary in order to enable coolant flow from lower part of the cavity to RPV wall. Main disadvantage of the approach applied for Loviisa plant with lowering the shield is the high cost of implementation and need to remove devices for ultrasonic inspection of RPV surface, which is located below reactor. On the other hand, the main benefit is the large thermal margin between expected and critical heat flux, what was experimentally verified on ULPU facility.

In Paks NPP the thermal/biological shield (Fig. 3.) will be not lowered, but a closable opening located in the central part of the shield will be installed. This buoyancy driven cover should open passively as soon as the water level in lower part of cavity reaches its elevation.

Through this opening water would flow into narrow curved gap between elliptical head and thermal shield. This is the critical point of ERVC loop, because the minimum gap and the maximum heat flux could be expected here during the accident. Due to the thermal expansion of the RPV under IVR conditions, the gap width would be reduced even further.

In order to support the implementation of necessary plant modifications for adoption of IVR concept licensing analyses have been performed.

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3 O. Kymäläinen, H. Tuomisto, T.G. Theofanous: In-vessel retention of corium at the Loviisa plant, Nuclear Engineering and Design 169, 1997
6. Supporting analyses

The analyses were made by IVS\textsuperscript{4} and IBOK\textsuperscript{5} Companies for Paks NPP. Two bounding severe accident sequences (LB and SB LOCA) without availability of HP and LP safety injection in power uprate (108 \%) conditions were carried out. In the 1\textsuperscript{st} sequence main safety concern was the maximum heat flux through the RPV wall. In the 2\textsuperscript{nd} sequence attention was paid to ability to depressurise the primary system using PORV and 1 (or 2) safety valve. The intentional reactor cavity flooding in the course of the accident was considered as a severe accident management measure in both sequences. This action will be included into SAMGs that are under development.

At first, integral analysis of both sequences was performed using the ASTEC code, which was jointly developed by IRSN and GRS. The most important operator interventions considered in the accident scenario were: depressurisation of secondary and primary system, draining of barbotage trays and flooding the reactor cavity.

As a next step, the obtained results were used as boundary conditions for structural analysis of reactor pressure vessel in order to obtain deformations of reactor outer surface. This analysis was performed with ANSYS code.

Finally, a detailed hydraulic nodalization was prepared for RELAP code considering the deformed shape of reactor outer surface and the analysis of ERVC was performed for the thermal load which was obtained by ASTEC code.

\textit{ASTEC results}

As it was mentioned, the thermal loading on the inner surface (time-dependent heat flux distribution through RPV wall) was estimated using ASTEC integral code (version V1.3 rev.2). Although the main concern was “in-vessel phenomena”, analysis of confinement response was also performed. In this analysis, made by IVS Company, the reactor cavity flooding is simulated applying appropriate boundary condition (heat transfer coefficient + coolant temperature) at the time when the coolant level in reactor cavity during flooding process reaches the RPV elevation. Main output from this analysis is the thermal loading. Other important outputs are:

- mass, composition and decay heat of molten corium pool in lower plenum;
- overall course of the accident, including confinement response.

In this approach the decay heat is calculated from the defined initial inventory of fission products (FPs) in the core. In the course of the accident, volatile FPs escape from the core (after cladding rupture). Non-volatile FPs together with molten fuel are relocated into lower plenum. This approach represents a realistic type of thermal load estimation, because basic corium pool parameters are estimated based on analytical estimation instead of postulating corium mass and decay heat in molten pool. Analyzed sequences were:

- LB LOCA (φ 200 mm) without availability of active ECCS;

\textsuperscript{4} P. Matejovic, M. Bachraty, M. Barnak: In-vessel corium retention for Paks NPP. Analysis of LB and SB LOCA sequences, IVS reports, Trnava, August 2009

\textsuperscript{5} R. Berky, J. Bosansky: Computational Analysis of Reactor Pressure Vessel Dilatation, IBOK Technical Report, Bratislava, August 2009
SB LOCA (ϕ 10 mm) without availability of active ECCS, with loss of offsite power.

The first sequence represents the largest LB LOCA, which does not result in spilling of coolant from barbotage trays. LB LOCAs without active ECCS injection are characterized by rapid depressurisation of the primary system at the beginning of the accident. Large coolant leak without availability of active ECCS results in early core heat-up, core melting, relocation and thermal attack of the molten corium on RPV wall. On the other hand, there is only limited amount of coolant available on the confinement floor. Thus, manual draining of barbotage trays is necessary in order to assure sufficient coolant volume for reactor cavity flooding, which takes relatively long time. This combination of early core melting and late reactor cavity flooding represents the most challenging sequence from the point of view of preserving RPV integrity. The main goal of this analysis is to prove that the flooding of the reactor cavity occurs sufficiently early for preserving RPV integrity. Heat flux profiles at different times obtained from ASTEC analyses for this LB LOCA case are shown on Fig. 4.

Fig. 4. LOCA 200 mm. Heat flux at RPV outer surface at different times (x axis represents spread distance from RPV bottom to the top along the surface)

The second accident represents a high-pressure sequence. For successful application of the IVR strategy the primary system has to be depressurised manually by operator via pressuriser safety and relieve valves to a sufficiently low level (p1 < 20 bar). Again, manual draining of barbotage trays is necessary to assure sufficient coolant volume for reactor cavity flooding. The main goal of this analysis was to prove whether it is possible to depressurise primary system and to flood the reactor cavity before the internal surface is subjected to thermal load from molten corium relocated into lower plenum. Heat flux profiles at different times for SB LOCA are shown on Fig. 5. Due to late core melting the maximum heat flux values are significantly lower than in the previous case.

ANSYS results

The obtained time-dependent heat flux distribution through the reactor inner surface and primary pressure were used in the subsequent structural analysis, which was performed using finite element code ANSYS (release 10). The prescribed boundary conditions were applied at both, inner (heat flux profile) and outer
(HTC and coolant temperature) reactor surface. Main output from this analysis, made by IBOK Company, was the time-depended deformation of the reactor outer surface.

The results of analysis of RPV dilatation taking into account stress relaxation due to creep are:

- The radial dilatation of the cylindrical shell of RPV in LOCA 200 mm scenario achieves the max. value 10.5 mm on the outer surface in the area of the interface between the cylindrical shell and head, at about 7.75 hours from the start of the accident.

- The radial dilatation of the cylindrical part of RPV at the LOCA 10 mm scenario achieves the max. value of 7.5 mm, at about 36 hours from the start of the accident.

Results of ANSYS analyses for the analysed LB LOCA case are shown on Fig. 6. and 7.

Fig. 5. LOCA 10 mm. Heat flux on RPV outer surface depending on time
Fig. 6. LOCA 200 mm. Thermal field in RPV with visualisation of melted zone, time of 7.5 h

Fig. 7. LOCA 200 mm. Radial displacement change on outer surface of RPV head during 1. day

**RELAP results**

In the last step, the analysis of external reactor vessel cooling (ERVC) was performed using the RELAP 5 code. Based on the information from ANSYS analyses, hydraulic nodalization was prepared considering actual deformed “hot-state geometry”. The reactor wall was modelled as heat structure heated on inner surface (applied thermal load from ASTEC analysis) and cooled on outer surface (actual convective heat...
transfer coefficients calculated by the RELAP code based on the actual two-phase flow conditions in control volume of ERVC loop, which is adjacent to given mesh point of heated structure).

The gap width between RPV outer surface and thermal/biological shield is locally decreased to minimum value ~10 mm for LOCA 200 mm and ~13 mm for LOCA 10 mm. Main goal of the ERVC analysis was to demonstrate, that under the IVR conditions there is no boiling crisis on the RPV outer surface and thus the integrity of RPV is preserved. Mass-flow-rate with oscillated shape through the ERVC loop for LB LOCA case is shown on Fig. 8.

From the results of RELAP analysis it follows that the above-mentioned gap width is sufficient for natural circulation of coolant in two-phase flow regime. In general, oscillatory flow behaviour was predicted in ERVC loop. Potential danger for RPV cooling could arise from short periods of flow stagnation, when higher voiding in the narrow gap between RPV wall and thermal/biological shield may occur. However, the heat transfer mode from RPV outer surface remains in subcooled single-phase convection (inlet – curved part of the riser) or nucleate boiling mode (rest of the riser). Thus, the boiling crisis was not predicted and there were only minor fluctuations of RPV surface temperature.

Efficiency of the ERVC loop in given IVR geometry should be proven experimentally by AEKI on CERES facility, which is under construction in Hungary.

![Fig. 8. LOCA 200 mm. Coolant mass flow rate in the riser](image)

7. Conclusions

An IVR concept with simple ECVR loop based only on minor modifications of existing plant technology was proposed for Paks NPP. The analyses supported the assumption that the proposed solution is effective in preserving RPV integrity in the case of severe accident. Possible uncertainties in code predictions are covered by the applied conservative assumptions.
Based on these encouraging results licensing document for implementation of necessary plant modifications was prepared. This engineering design can be characterised as passive, low costs and easy implementing one. This solution after having licence from regulatory body is planned to be implemented in 2011 on the 1st Unit in Paks NPP.

**Abbreviations**

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<tr>
<th>Acronym</th>
<th>Description</th>
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<tr>
<td>ECCS</td>
<td>Emergency Core Cooling System</td>
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<td>ERVC</td>
<td>External Reactor Vessel cooling</td>
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<td>CERES</td>
<td>Cooling Effectiveness on Reactor External Surface</td>
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<td>FPs</td>
<td>Fission Products</td>
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<td>HTC</td>
<td>Heat Transfer Coefficient</td>
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<td>IVR</td>
<td>In-Vessel Retention</td>
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<td>PORV</td>
<td>Pressuriser Relief Valve</td>
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<tr>
<td>RPV</td>
<td>Reactor Pressure Vessel</td>
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<tr>
<td>SAM</td>
<td>Severe Accident Management</td>
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<td>SAMG</td>
<td>Severe Accident Management Guidance</td>
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Introduction

- Paks NPP: 4 units VVER-440/V213 type with bubble condenser
- safety upgrading program completed -> all modifications to increase capability to cope with DBAs (with power uprate)
- long term goal: life time extension -> SAM program developed
- IVR via ERVC: promising SAM strategy for existing plants
- additional technical modifications, necessary for ERVC loop is rather limited, but compensated with significantly lower maximum heat fluxes
- to prevent RPV failure: cooling molten core internally or RPV wall externally?
  - RPV: relatively high surface area <-> low decay power
  - external cooling more effective

Upgrading of Paks NPP for severe accidents

Key elements of the SAM strategies

- Prevention of core damage
- In-vessel retention or exvessel debris cooling
- Release and containment management
Upgrading of Paks NPP for severe accidents
AM strategies and their components

- Measures to prevent core damage:
  - Strictly perform the adequate EOPs to prevent severe accidents

- Measures to prevent RPV failure:
  - Primary circuit depressurization by opening of PRZ valves (in EOPISAMG)
  - In-vessel corium retention by cavity flooding (in SAMG) ⇒ to safe cavity integrity in ex-vessel case - not challenged

- Measures to safe confinement integrity:
  - Hydrogen treatment: application of 30 large passive recombiners
  - Prevention of late over-pressurization by filtered venting (in SAMG) ⇒ modification of existing confinement vent system

In-vessel retention for VVER-440/V213 units
Related design features

+ low core thermal power;
+ large water volumes in primary and secondary systems (+ 1000 m³ on bubble condenser trays);
+ RPV without penetration in lower head;
+ high RPV ⇒ high driving head for natural circulation in ERVC loop;
+ large diameter ventilation ducts of cavity cooling system;
+ annulus gap (width ~30 cm) between RPV and thermal insulation of cylindrical part of RPV (riser of ERCV loop);
  - presence of massive thermal/biological shield of lower head;
  - flow restriction for steam-water mixture venting at the outlet from the cavity to confinement
In-vessel retention for VVER-440/V213 units
Related design features

Two proposals of IVR: with lowering of thermal/biological shield (left) and with central hole in it (right)

Critical points of the ERVC loop:
- vertical gap ~3 cm due to presence of RPV support structures
- vertical gap on the elevation of biological shield only ~2 cm, but max. heat flux expected here!
- thermal expansion of the RPV under IVR conditions => gap width reduced below 2 cm!
IVR proposal for Paks NPP

Layout of the ventilation ducts for cavity cooling, Paks NPP

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IVR proposal for Paks NPP

Modification of ventilation ducts TL03 (isolation siphon) Passive opening (buoyancy driven cover) on the thermal shield

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IVR proposal for Paks NPP

Thermal/biological shield of RPV elliptical head, Paks NPP

Supporting analyses

- Analyses made by IVS and IBOK Companies (Slovakia)
- 2 bounding severe accident sequences (LB and SB LOCA) without HP and LP injection in power uprate (108 %) conditions, with operator actions in EOPISAMG:
  - LBLOCA is most challenging sequence: maximum heat flux through the RPV wall, but no spilling of water from barbotage trays ⇒ manual draining (70-90 min.)
  - early core melting - late cavity flooding
  - SBLOCA is high pressure sequence: primary depressurisation using PRZ valves (p,< 20 bar)
- Integral analysis of both sequences using ASTEC (V1.3 rev.2) code ⇒
- Structural analyses using ANSYS code:
  obtained by ASTEC results (time-dependent heat flux distribution through the reactor inner surface and primary pressure) used as boundary conditions ⇒
- Analysis of ERVC loop performed using RELAP 5 code:
  deformed shape of RVP outer surface (min. gap) from ANSYS and termal load from ASTEC calc. ⇒ boiling crisis should not expected
Supporting analyses
ASTEC results

LOCA 200 mm: Distribution of heat flux at RPV outer surface depending on time

LOCA 10 mm: Distribution of heat flux at RPV outer surface depending on time
Supporting analyses

**ANSYS results**

LOCA 200 mm: Thermal field in RPV with visualisation of melted zone, after of 7.5 h

⇒ Gap width between RPV outer surface and thermal/biological shield is locally decreased to minimum value:
- ~10 mm for LBLOCA case
- ~13 mm for SBLOCA case

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**Supporting analyses**

**RELAP results**

LOCA 200 mm: Coolant mass flow rate in the riser - boiling crisis not predicted!

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Conclusions

- IVR concept with simple ERVC loop proposed for Paks NPP.
- Analyses supported: proposed solution is effective in preserving RPV integrity in case of severe accident.
- Licensing design documentation for implementation of necessary plant modifications prepared.
- Engineering design: mostly passive, low costs and easy implementing.
- Efficiency of the ERVC loop in given IVR geometry will be proven experimentally by AEKI on CERES facility (under construction).
- After having licence from regulatory body is planned to be implemented in 2011 on the 1st Unit in Paks NPP.