Safety of the Mochovce NPP in comparison with Western safety requirements.

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1 Introduction

In 1993, the Slovakians decided to complete the construction of the Mochovce NPP in line with Western standards.

The fulfilment of this decision has required a comprehensive modernisation programme, because the construction of four units of the Soviet-designed advanced VVER-440 of Mochovce has already started in 1984. After political changes, construction was inter-rupted due to uncertainties in connection with the future economic developments and concerns with regard to safety requirements. After the revision of the decision, the first two units Mochovce-1,2 were to be completed with Western financial support through credits from the European Bank for Reconstruction and Development (EBRD) and the European Investment Bank (EIB). On behalf of the European Union (EU), within the framework of PHARE, IPSN and GRS made a review in 1994 of the Industrial Modernisation Programme taking into account the status of plant equipment subsequent to the long-term interruption of construction. The main task of the review was to see whether the upgrading measures proposed in the modernisation programme were adequate to fulfil Western requirements.

In early 1995, the Slovak government decided not to take credits from the EBRD and the EIB and to finalise units Mochovce-1,2 by own national efforts with participation of the Czech, Slovak, Russian and Western Industries. Both the utility Slovenske Elektrarne (SE) and the Slovak Safety Authority (UJD) confirmed that all upgrading measures for achieving Western safety standard would be realised and that in particular all recommendations given by Riskaudit and IAEA would be fulfilled. For assisting the Slovak Safety Authority, a second TSO PHARE project was initiated in 1998. Participants of this TSO project under the leadership of Riskaudit have been the Italian ANPA, Spanish CIEMAT, German GRS, French IPSN and Hungarian KFKI.
The main tasks of the project have been the following:

Review of upgrading measures planned to be implemented, Review of safety documentation on the basis of Western practice,

Review of commissioning programme and main results of the nuclear tests,

Assessment of the overall safety status of the plant.

1 Basis of Evaluation

The overall aim of this project was to check whether the Mochovce NPP fulfils the safety requirements and safety practice applicable for Western operating NPPs. It was agreed with UJD that the recommendations formally given by Riskaudit and the IAEA in the Safety Issue Book would be considered as a reference for the safety review. Current international recommendations together with elaborated guidelines have been used for performing the safety review. The main documents used for this purpose were several INSAG documents, such as INSAG-3,-6,-8,-10 and the IAEA Series No. 12.
In addition, the participating Technical Support Organisations (TSOs) made use, when applicable, of insights from their own national regulations and applied practice.

The licensing approach of the Slovak Safety Authority is based on deterministic analyses. A standard format of the Safety Analysis Report (SAR - according to RG 1.70 US NRC) was agreed with the Mochovce NPP in 1996.
Complementary to the deterministic analyses, the UJD has also requested to fulfil a probabilistic target according to international standards 10^{-5} CDF per year for the overall safety status of the plant and confirmation that the risk profile is well balanced, e.g. with no single initiating event, accident sequence, system failures or personnel errors contributing more than about 10% of the total Core Damage Frequency (CDF) or the frequency of severe off-site releases.

Additionally, the UJD requested to perform analyses of accidents occurring during shutdown states and Beyond design Basis Accidents (BDBA - according to IAEA guidelines IAEA-EBP-WWER-09, July 1997).
For the evaluation of the safety of the Mochovce NPP, the defence-in-depth concept was applied. It was verified whether

- the design provides appropriate defence in depth on all levels,
- the risk of intolerable consequences for the plant, humans and the environment is sufficiently low,
- all factors essentially contributing to safety have been assessed according to applicable current standards and practices.

Within our evaluation, the main emphasis was on the assessment of measures preventing accidents (levels 1 and 2) and, in the case of accidents, of measures to cope with them in order to prevent more severe conditions jeopardising the concept of radioactivity barriers (levels 3 and 4).
The approach used was based mainly on the deterministic basis, following the recommendations given by the IAEA, NUSS and INSAG documents. Referring to the various safety levels, special attention was given to selected issues.
On the first level of defence in depth, a combination of conservative design, quality assurance (QA), surveillance activities, operator qualification and training, equipment qualification, the basic safety design of components and their integrity,

status of in-service inspection, commissioning programme and test results were considered.
On level 2, some examples of control devices were examined in order to verify the plant's response to abnormal operating conditions or any indication of system failure.
On level 3, several steps have been taken, such as:

- verification of the completeness of the list of postulated initiating events (PIE) according to the corresponding IAEA guideline, site-specific requirements for the Mochovce NPP, and operating experience feedback from other VVER-440 reactors,
- verification of the PIE classification in terms of frequency/consequences, along with acceptance criteria,
- evaluation of the computer codes used in view of their qualification for VVER specifics,
- verification whether the fundamental safety functions can be ensured by the safety-relevant systems, components and structures
  - control of reactivity
  - core cooling
  - enclosure of radioactive materials,
- verification whether deterministic requirements were met for the safety systems. The main requirements are as follows:
  - single-failure criterion
  - common-cause failure (CCF) including loads duo to general hazards,
  - physical separation
  - functional diversity
- verification whether adequate information and sufficient time for intervention is available for accident sequences requiring operator intervention.

On level 4, some selected beyond-design-basis sequences have been examined in order to verify the prevention of core degradation for the following initiating events:

- ATWS,
- total blackout,
- loss of main and emergency steam generator feedwater supplies.

Attention was paid also by the reviewers to the tendencies coming mainly from the probabilistic studies carried out for existing Western plants and from their operating experience. They are related to the sequences leading to core melt

- under high pressure,
- with containment bypass and
- in shutdown conditions.

The compliance of the UJD probabilistic safety target could not be assessed, since the PSA-1 reflecting the real status of the plant including the upgrading measures and sequences in shutdown states has not yet been completed.
But we could review the first part of PSA reflecting the pre-modification status (before implementation of upgrading measures). It shows, that the PSA methodology, techniques and models correspond to the present state-of-the-art

and that plant features are reflected in the PSA model in an adequate manner. The utility concluded for the total CDF without and with Feed & Bleed procedure 1 x 10^{-3} per year unit and 6.4 x 10^{-5} per year unit respectively.

2 Specific issues

Starting the project on Mochovce, the following relevant issues for VVER-440/213 reactors were discussed:

1. Integrity of the reactor pressure vessel (RPV),
2. Integrity and tightness of the Bubble Condenser Containment (BCC),
3. Ensuring of emergency core cooling re-circulation via the containment sump,
4. Vulnerability of the emergency feedwater system (EFS),
5. SG primary header lid,
6. Reliability of I&C,
7. Scope and methodology of accident analysis (AA),
8. Seismic design,
9. Commissioning.

2.1 Integrity of the reactor pressure vessel

The integrity of the reactor pressure vessel has been examined extensively by various groups of experts, keeping in mind that the general design of the VVER 440 results into a fairly high neutron flux at the vessel wall. Safety concerns were raised regarding:

- sufficient knowledge of the «beginning of life» material properties,
- the adequacy of the surveillance programme and
- the enveloping character of the transients selected for the thermal-hydraulic and integrity analysis.

Furthermore, the as-built status regarding the absence of non-acceptable defects verified by non-destructive testing and the absence of detrimental attack by the long conservation due to the break in construction had to be assessed. A further relevant topic was the reduction of the neutron flux at the pressure vessel wall since the beginning of the operation as previously recommended.
The results of the detailed common discussions can be summarised as follows:

- The methodology used in the integrity assessment of the reactor pressure vessel, based on the Russian normative procedures, are conservative in principal because the assessment is performed with the goal to exclude crack initiation. This methodology is conservative compared with methodologies contained in some of the Western codes and standards because it does not take into account warm pre-stress effects and does not allow any kind of crack extension.
- The rules applied for in-service inspection of the reactor pressure vessel requesting 100 % inspection of the welds at a shorter time interval compared with most of the Western standards.

- Based on the well-documented material properties from the fabrication, the «end of life» transition temperature is estimated to be well below 100 °C (estimated value 71 °C), which is well below actual licensing criteria applied in many Western countries.
- Available analyses showed sufficient margins for cracks in circumferential direction. Cracks in axial direction have smaller margins, but they are unimportant because of the higher ductility of the base metal on both sides of the weld.

- Taking into consideration these technical aspects and the implemented prevention measures against PTS loads, the delay of a low-leakage core management to the next fuel cycles is judged to be acceptable.

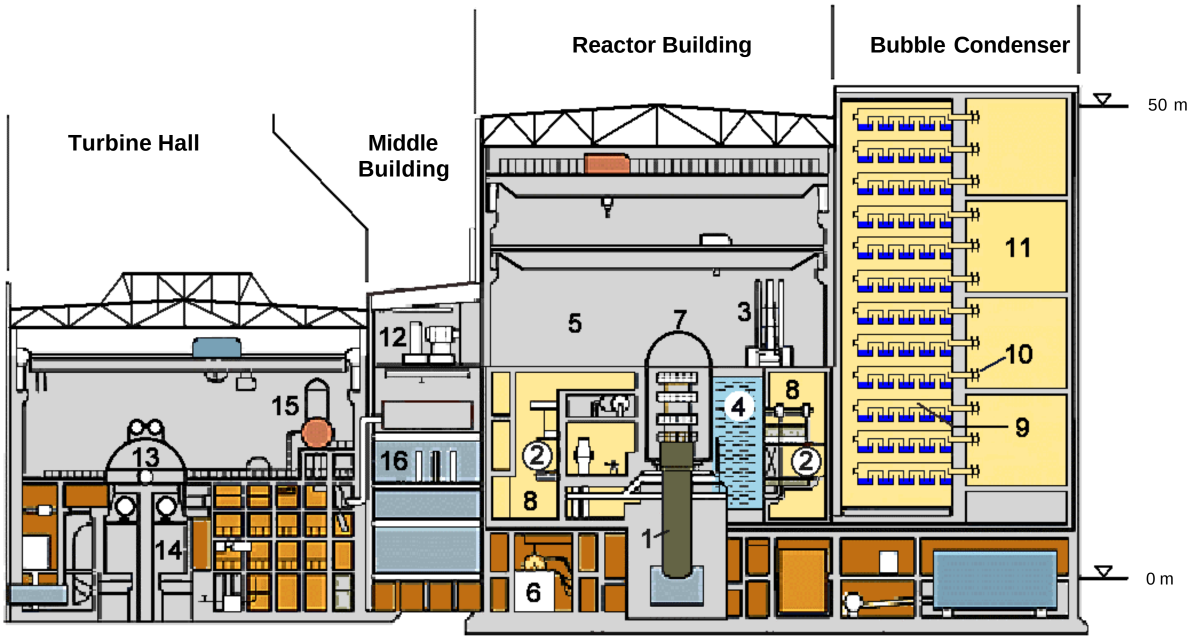
Reducing PTS loads by heating the ECCS water up to 55 °C, other protections are implemented against cold over-pressurisation and sub-cooling in case of steam line rupture.

2.2 Integrity and tightness of the Bubble Condenser Containment

The integrity and tightness of the Bubble Condenser Containment (BCC) have been examined, recognising that the general design of the VVER-440/V213 has a specific containment construction with a steam condensation system for pressure suppression.

Sectional plane of the Bubble Condenser Containment

Safety concerns were raised regarding sufficient safety demonstrations of:

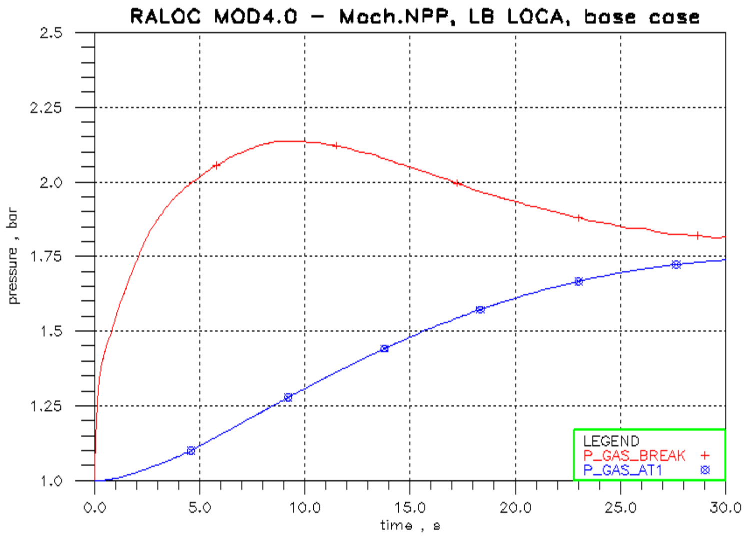


1. Reactor pressure vessel, 2. Steam generator, 3. Refueling machine, 4. Spent fuel pit, 5. Reactor hall, 6. Make-up feedwater system, 7. Protective cover, 8. Confinement system, 9. Bubble condenser trays, 10. Check valves 11. Air traps, 12. Intake air unit, 13. Turbine, 14. Condenser, 15. Feedwater tank with degasifier, 16. Electrical instrumentation and control compartments

- the maximum pressure in the containment and pressure differences between compartments build-up during a rupture of a main cooling pipe (large break LOCA),
- the fluid structure interactions during different LOCAs, such as oscillations caused by condensation and chugging caused by air-free steam condensation within the water pools.

Furthermore, the loss of components important for pressure suppression during LOCAs, such as gap caps, water trays and check valves, had to be assessed.
The results of the review can be summarised as follows:
The maximum pressure expected to occur in the steam generator compartment after about 10 s depends mainly on the mass and energy release rate from the primary

circuit, the hydraulic resistances along the flow path from the hermetic compartment system, through the Bubble Condenser (BC) to the air traps. For the reference case without consideration of any failure regarding the BC, the maximum pressure will be well below the BCC design pressure. Former calculations have not shown sufficient safety margins mainly due to an overestimated reactor coolant mass and energy release rate adjoined with the LB LOCA (160 GJ instead 120 GJ)

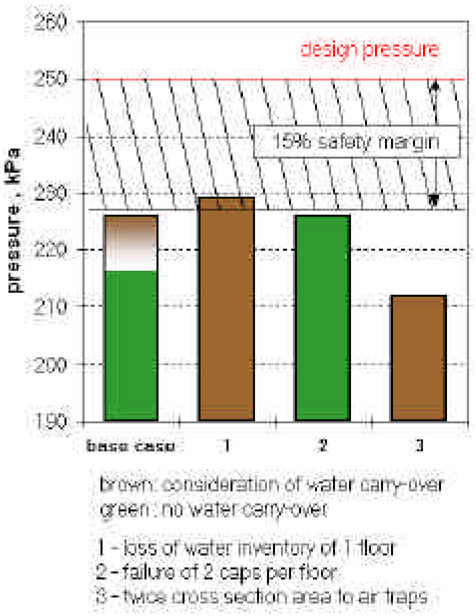


For assessing the pressure suppression it is necessary to follow the path of the steam-air-mixture from the leak to the air traps:

- two channels connecting the steam generator box and the BC gas shaft (50 m^2),
- 1800 gap caps for steam condensation with a cross-section area of about 160 m^2
- 12 check valves DN500 to the air traps with a total cross section area of 2,36 m^2.

The cross sections provide hydraulic resistance with significant impact to the maximum pressure build-up. The modelling of the flow through the check valves, which have the narrowest cross section, showed their dominated contribution to the maximum calculated pressure peak in the BCC. The water carry-over is sensitive to the resistance of the valves.
Tests with original valves (two in series), performed in Karlstein, have confirmed the assumptions used for the pressure analyses and have shown in addition that the effect of water carry-over is not so significant.
The calculated maximum pressure in the BCC increases under consideration of water carry over by about 8 %. Contrary to that, the doubled cross section area leading to the air traps would decrease the maximum pressure by about 11 %.

Furthermore, the bypass of the water layer, caused by the loss of a certain number of caps or of emptied trays, was also considered. The condensation efficiency of the whole BCC almost remained with regard to the pressure suppression efficiency. This is due to the self-compensation properties. A bypass in one BC section leads to a fast pressurisation of the corresponding air trap. Consequently, the counter-pressure leads to flow limitation compensated by a flow increase and then to a more efficient condensation of the non affected BC sections.



The integrity and the tightness of the containment were demonstrated by the design overpressure test, about 30 % higher than the maximum accident overpressure. The measured leak tightness of 2 %/day gives indication about good construction quality. It is the best result of all VVER-440 so far.

The integrity of tray levels 1 to 11 and the gap cap walls has been analytically and experimentally demonstrated up to pressure differences of 30 kPa, which is a very conservative value as compared to the thermal-hydraulic analysis results.

The analysis, verified by experiments simulating different pressure gradients in the gas-shaft, showed that the maximum pressure difference produced during the first second of an LB LOCA is less than 20 kPa. Some plastic deformations were produced only at 30 kPa, but the structure remained operable.

Additionally, the following reinforcements of the gap cap systems and BC walls made in Units 1 and 2 improve the structural behaviour of the BC system:

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- Lateral support beams on the first and eleventh tray levels.
- Spacers inserted between gap caps and ceiling stiffeners to improve the transmission of horizontal loads from the lateral support beam (see the right picture).
- Columns were added between the first and second tray level support beams.

The only weak point of the BC system are the walls of tray number 12 due to its enlarged height (4.2 m instead of 2.7 m). However, the loss of the integrity of this has no significant impact on the maximum BCC pressure peak. Nevertheless, it would be recommendable to reduce the differential pressure across the wall of this tray, e.g. by reducing the water level inside the tray.

The fluid structure interactions tests with two half gap cap sections performed in Praque-Bechovice/Czech Republic have not detected any dangerous oscillations caused by condensation and chugging caused by air-free steam condensation within the water trays. The results show that oscillations occurred with a maximum pressure difference of approximately 5 kPa across the caps. The frequency of these oscillations is 2 Hz and the load is maintained for just 20 - 25 seconds. The stimulated oscillation does not reach the basic eigen-frequency of the gap cap walls, which is around 4 – 5 Hz.
The results confirm also the specific chugging reducing construction of the gap cap system with a very extended cross section area – 1800 m long and 9 cm high - zig-zag shaped lower edges of the caps and a small insertion depth of 0.5 m through which the steam is injected into the water layer. However, main steam line breaks (MSLBs) and small break (SB) LOCAs have not yet been investigated during these tests.

Based on analyses, a hydrogen control system consisting of 8 hydrogen monitoring sensors and 16 passive autocatalytic recombiners was installed at the BCC. The recombiners are capable of limiting the hydrogen concentration in the compartment system < 2.2 Vol %. This is well below the critical concentration of 4 Vol %.
Taking the investigated technical aspects into consideration, the integrity of the third barrier during LB LOCAs will be maintained. The final confirmation is expected by the end of 1999 as a result of the Phare/Tacis Project "Bubble Condenser Experimental Qualification". Especially the experiments being currently performed on the large-scale test facility at EREC Electrogorsk/Russia will serve to:

- confirm BCC functionality during LB LOCAs as well as under MSLB and SB LOCA conditions,
- make it possible to verify applied calculation tools.

Considering all the analytical and experimental work performed, the BCC of the Mochovce NPP is presently the best investigated containment of a VVER-440/V213 type of reactor.

2.3 Ensuring long-term emergency core cooling by re-circulation of the primary water via the containment sump

Following a loss-of-coolant accident (LOCA), the water discharged from the break reaches the containment sumps during the re-circulation mode. Successful long-term re-circulation depends upon the level of clogging of the sumps. Moreover, in the re-circulation mode, pumps - high- and low-pressure safety injection and containment spray - have to carry water, free of debris and air, at a sufficient pressure in order to satisfy the net positive suction head requirements.

The following three issues have been examined:

- generation and the transport of debris,
- sump screen clogging,

- pump performance, taking into account the head losses of the circuit.

The quantity and segmentation of the debris were defined by experiments based on the degradation of two kinds of thermal insulation in the Russian KASHIRA Institute. Based on these results and the defined break site of the main cooling pipe in the vicinity of the steam generator, 1100 kg of insulation with a mean thickness of 160 mm and a surface of 40 m² would be dislodged (this quantity of insulation is assumed as a conservative approach, duo to the implemented LBB concept for the main cooling pipes and leak detection measures for the high energy secondary pipes). About half of insulation would be carried to the sump, part of it crumbled in fine particles. The other insulation is supposed to be sinking. Using these parameters, the utility has designed the screens. The screens are grids (2 x 2 mm mesh size) which have an angle of 20° with the vertical. This slope is presented as a passive means to clean them. Furthermore, trash rakes (100 x 100 mm) are located around the screens. Their overall surface (about 14.5 m^2) is determined for a filter velocity in compliance with the ECCS flow rate during LB LOCAs.

The behaviour of the new designed screens was verified on the small-scale facility in VUEZ/Slovakia. The maximum loss of pressure was observed with sufficient margin under the design value during a 72-hour test. The results give the possibility for monitoring screen clogging and strategies in case of unacceptable clogging detected by the monitoring system. Therefore two independent level measurements, each in front of and behind the screen and based on different physical principles, are implemented (conductivity and immersion pressure measurements).

Even if the corresponding procedures need some more justification, the reviewers conclude that the plant has successfully demonstrated the long-term operability of the ECCS even in the case of an accumulation of all generated debris in one sump. Active cleaning of the screens is not needed. The choice of heat insulation, Nerofil, was proven, and a replacement is not necessary. The lessons can be learnt by other plants to improve their long-term core cooling reliability.

2.4 Vulnerability of the emergency feedwater system

An essential deficiency of the VVER design was eliminated by new pipe routing of the emergency steam generator feedwater system (EFWS). Now the pipes are physically separated out of the turbine hall and elevation 14.7 m and are thus protected from common-mode failures such as fire or consequential failures such as the rupture of high-energy steam or feedwater pipes or turbine missiles (see picture).

The new lines are seismically protected and connect the emergency feedwater building outside the turbine hall with the steam generator compartment through the air trap of the containment and Bubble Condenser.

The EFWS storage tanks can be re-supplied by diverse means. Nevertheless, nozzles are implemented to connect fire tracks to the emergency feedwater pipes, and preventive accident management procedures have been established.

The implemented measures are in compliance with Western practice.

Secondary and primary feed and bleed

The total loss of the steam generator feedwater systems can lead to core meltdown under pressure. The total loss of feedwater happens after the loss of normal, auxiliary and emergency steam generator feedwater and further after emergency steam generator supply from fire trucks. Consequently, primary and secondary feed and bleed as well as a combination of both have been investigated. Feed and bleed is covered in the Westinghouse procedures as well as by the current event-oriented procedures. There is an about 40-minute period when the set-point for emergency systems «low steam generator level» is reached. If the heat removal could not re-established, then there is another five hours left during which the secondary feed and bleed procedure with emergency feedwater pumps and steam dump stations or, as the last and ultimate means in the emergency procedures, the primary feed and bleed should be applied.
The results of the primary feed and bleed analyses showed that by the combined action of the bleed valves for depressurisation and the HPSI pumps for make-up of the primary inventory, primary feed and bleed prevents inadmissible core uncovery down to low primary pressures where residual-heat removal conditions are reached.
One pressuriser safety valve is the minimum bleed valve configuration, which ensures:

- adequate core cooling during the phase of depressurisation and a reliable transition phase to residual-heat removal conditions,
- sufficiently low bleed rates to minimise the transient temperature and pressure load upon the vessel,
- always sufficient margins of the P/T curve of the plant to the brittle fracture curve,
- in the long run, low steady-state pressure and temperature values.

Concerning the pressuriser valves, a pilot valve has been added to allow the manual opening of the relief valve from the control room. It has been claimed that also the safety valves are qualified to carry water-steam mixture or water. Nevertheless, some aspects of the qualification need further discussion.
Primary feed and bleed is considered as the last and ultimate means in emergency operating procedures to prevent core damage. The application of primary feed and bleed procedure intervenes, in case of secondary-side heat removal unavailability, when the core exit temperature reaches 370 °C.
Even if all thermal-hydraulic studies have not been reviewed in depth, it can be concluded that the principle of the proposal made by the utility is satisfactory and positive.

2.5 Cover lift-up of the steam generator primary header

Initiating events in connection with large leaks from the primary to the secondary side (PRISE) of the steam generator, if the safety injection inventory is not enough, could turn over to a severe situation with containment bypass. To avoid this, several modifications of the SG primary header were implemented. They are aimed to exclude an opening of the cover or as well as to limit the leak < DN 42 (postulated opening cross section):

- new reinforced primary header lid with a stop, limiting the cross section for opening,
- new sealing rings made of expanded graphite,
- new bolts and washers for better distribution of tensions,
- new technological procedure of the lid assembly using a hydraulic set to strain and tighten the bolts,
- reinforcement of the secondary shell of the steam generator to limit the impact of the collector cover in case of lift-up,
- in-service inspections and leak control.

The primary collector header leak was analysed using CATHARE computer code. The upgrading measures were considered. Calculation was stopped when the leak stopped too, i.e. when primary pressure was equal to the secondary pressure of the affected SG.

It appears that the operation of one steam dump station (BRU-A) is sufficient to avoid the opening of the SG safety valves. The study has been performed in two cases : at nominal and hot zero power. In the first case, the SG is filled at 600 s, in the second case at 200 s. Consequently, it has been decided to use qualified steam dump stations to carry water and steam water mixture, and not to re-qualify SG safety valves.

The other improvements to cope with PRISE are the following:

- installation of N16 activity sensors on the steam lines,
- regulation of the flow rate of the high-pressure safety injection (HPSI),
- emergency procedure for switching off the high-pressure safety injection pumps 10 minutes after the initiation of the accident (however, in the accident analysis, the operator does not intervene before 30 minutes),
- qualification of the steam dump stations to carry water and steam-water mixture,
- implementation of a second pressuriser water spray line.

In the accident analysis, the contribution of the main isolation valves during the accident was not considered. Nevertheless, in the emergency operating procedure, the possibility to use these valves has been introduced as an alternative way. Furthermore, a procedure exists to prevent dilution by back-flow from the secondary side to the primary side by stopping the pressuriser spray at 2 bar.

In the safety report (POSAR), a limiting case of PRISE of equivalent diameter equal to 42 mm has been analysed. Basically, conservative assumptions were used in the analysis, but the single failure application - loss of one auxiliary spray train - is questionable. As a matter of fact, each steam generator steam line has two steam dump stations (BRU-A). One steam dump station has an automatic

opening mode and the another one a manual mode. The application of the single failure criterion to the first steam dump station might lead to a worse scenario since the steam generator safety valves, not qualified to discharge steam water mixture, would be challenged during the first 30 minutes when no credit can be given to the second steam dump station manually operated.
According to the calculated radiological consequences, the selected scenario PRISE (DN 42) leads to radiological consequences in compliance with DBA radiological targets.
Even if the chosen PRISE scenario needs further justification with regard to the single failure application, the reviewers assess the situation to be in good progress. They conclude that the performed improvements contribute to enhance the unit's safety.

2.6 I&C of safety systems

Safety concerns were raised among others regarding:

- reliability of the I&C of safety systems,
- human engineering of control rooms,
- insufficiency of prevention measures such as diagnostic and accident monitoring.

In the following, the main results of the review, especially according to the compliance with international criteria and interactions between the new installed systems and the remaining parts of the original system will be presented:

The design of the reactor trip system (RTS made by Russian SNIIP) and the engineered safety feature actuation system (ESFAS made by Siemens) were significantly renewed comparing with the original I&C design. Three reactor fast shut criteria were added, for $L^{+}_{pressuriser}$, $p^{+}_{primary}$ and $T^{+}_{hot\ legs}$, which the

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Emergency control room - ESFAS panel

Russian designer Gidropress considered unnecessary for the safety of VVER-440.
The RTS consists of two physically

separate sets of three channels each. The single-failure criterion in case of repair or testing during operation will be fulfilled. It improves the reliability significantly compared with the original design. The presented results of the unavailability of one or all three trains of RTS and ESFAS on demand with consideration of common-cause failure are well below the requirements for the basic design of the Mochovce NPP. The reactor protection system – RTS and ESFAS - are completely qualified and proofed according Western standards.

Furthermore the safety prevention was strengthened by re-designed normal operation I&C systems, as plant computer, control units and control rooms, according to international standards (ISO 9000, IEC, KTA, RCCE). Beside the improved reliability of I&C systems and well performed personnel training, ergonomic improvements also contribute to a decreased probability of man-induced failures, e.g. the main and emergency control rooms are rearranged according to Western ergonomic practice.

Main control room

In connection with the goal to meet the leak-before-break criterion, three diagnostic leak detection systems have been installed. They work on diversified physical principles – acoustic emission, activity measurement in the immediate vicinity of the components, and humanity monitoring. The measurements of all relevant systems are analysed by a special computer system –CS LBB- in order to provide transparent information for decision-making in case of a leak. Furthermore, some other newly installed diagnostic systems serve for early detection of loose particles, equipment vibrations, movements of primary circuit components, and fatigue monitoring. A leak detection system is also installed on the main feedwater and steam lines of the secondary circuit.
Taking into account some remaining measures the completion of which will be finalised during next year, such as:

- completion of the post-accident monitoring system,
- completion of the independent ventilation systems of the main and emergency control rooms,
- completion of the reliability analyses of the reactor trip system with regard to:
  - common-cause failures initiated by design or manufacturing faults,
  - erroneous personnel action in adjusting set-points of the neutron-flux power limitation,

there are no major obstacles for achieving the safety level in compliance with Western I&C practice.

2.7 Scope and methodology of accident analysis

The main concerns were related to:

- scope and methodology of accident analyses,
- computer code and plant model validation,

- overcooling transients related to pressurised thermal shock of the RPV,
- steam generator collector rupture,
- accidents under low-pressure and shutdown conditions.

The results of the detailed common discussion can be summarised as follows:

- the completeness of the list of postulated initiating events (PIE) was verified according to the IAEA guide «Guideline for accident analysis of WWER NPP», site-specific demands for the NPP Mochovce and operating experience feedback from other VVER-440 reactors,
- the PIE were classified in terms of frequency/consequences, associated with acceptance criteria,
- the used calculation tools, as RELAP 5 and CATHARE 2, were evaluated in view of their qualification for VVER features,
- for accident sequences requiring operator intervention, it has been checked whether adequate information and sufficient time for intervention has been available,

The accident analyses have been performed with a deterministic approach to the occurrence of failures and functionality of the system and are adequately conservative.

· As an example, the inadvertent opening of the primary pressure safety valve followed by a delayed re-closure at the highest stress condition of the RPV was selected as the most critical PTS event from the thermal-hydraulic point of view. For this PTS event, structural analyses were performed using the linear-elastic approach.
However, this event shows the most adverse results based on elastic calculations regarding structural response; it was recommended to complete the PTS analyses including the above-mentioned initial events, by using the elasto-plastic approach.
On level 4 some selected beyond design sequences have been examined in view of core melt prevention:

- ATWS,
- Station black out,
- Loss of main and emergency feed water supply,
- Loss of service water system.

Globally, it has been checked whether the dedicated measures (equipment and/or emergency procedures) ensure core melt prevention. Nevertheless, it will be necessary to examine in detail the list of BDBA in the light of PSA results. Also, the relevant thermal-hydraulic studies need further discussion.
The accident analysis complies with current international practice with regard to the scope of PIE’s, the failures analysed and the methodology used.

2.8 Seismic design

The original Mochovce NPP design took into account

- level IV of MSK-64 for SSE and horizontal peak ground acceleration HPGA = 0.06 g

· The seismic input parameters were re-evaluated in accordance with current international practice:

- level VII of MSK-64 for SSE which corresponds to the NUREG-0098 spectrum to minimum HPGA = 0.1 g and VPGA = 0.067 g.

For the confirmation of the new Design Basis Earthquake, a lot of studies have been performed in geology and seismology. Both deterministic and probabilistic seismic hazard approaches have been applied. They are based on seismotectonic zoning. The site is located inside a wide « aseismic » zone extending up to 50 km around the site, and in the case of the deterministic approach, earthquakes will not be moved closer than 50 km from the site.

However, the design value of HPGA = 0.1 g could not demonstrate a conservative margin. The critical point is the seismotectonic zoning with an exclusion area of 50 km.

Nevertheless, the applied methodology for the performed seismic re-evaluation was largely in accordance with Western practice:

- principles of the cool-down of the reactor after an earthquake,
- identification of the safety- related structures and safe shutdown equipment,
- evaluation of the seismic limit resistance using different methodologies and walk-downs or calculation of critical groups of components,
- comparison of seismic loads with response spectra of the location,
- evaluation of the anchoring,
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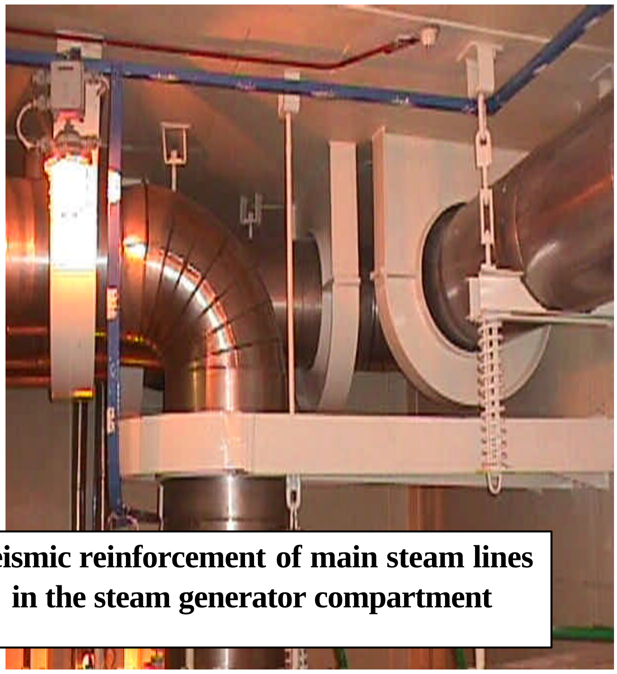
- identification of possible interaction.

In consequence of the performed re-evaluation, a lot of improvements and reinforcements were implemented for buildings, structures and components. Also, the seismic monitoring system was renewed.



Seismic reinforcement of main steam lines in the steam generator compartment

Based on preliminary investigations performed by Riskaudit and the review results of an IAEA mission, the UJD has taken a positive decision and requested the utility to re-evaluate the site related seismic characteristics. As a matter of fact the IAEA mission issued recommendations related to :

- the geological data base,
- the faulting in the site vicinity,
- the historical seismicity evaluation and attenuation to the site,
- the seismotectonic model,
- the modelling uncertainty,
- the response spectra.

Seismic reinforcement for Hydro-accumulators

Depending on the results of the site related characteristics, the NPP structures and components could be re-assessed for the Design Basis Earthquake to define and to implement their upgrading if required.

2.9 Commissioning tests programme and results

The commissioning tests programme and results were partially reviewed. As for reactor physics, it can be concluded that:

- the test programme is detailed and considers the essential features determining reactor behaviour and the safety-relevant aspects and indicates the acceptance criteria,
- the execution of tests and the evaluation of results is supported by well established report systems consisting of specification and documentation reports,
- the tests have been performed successfully,
- independent calculations which were performed by the Hungarian KFKI for some selected experiments confirm the measurements as well as the design basis calculations (see figure).

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Comparison of power distribution in the reactor core
(cycle 1)
First row : number of fuel element
Second row : measured value
Third row : calculated by KARATE-440

3 Main results of our evaluation

Summarising the results of our review so far, the following most important conclusions for the safety of the plant can be drawn:

The utility has taken into account all the issues addressed by the IAEA in 1996 and by Riskaudit in 1994. For their correction, measures were developed and implemented. In 1998, before the start-up of Mochovce-1, more than the half of all 89 issues were corrected, among them all issues with highest rank, category 3, except the not yet completed protection to high energy pipe breaks on 14,7 m platform. Another set of about 25 issues, among them almost all issues of category II, should be corrected before the second fuel cycle will start in December 1999. The remaining part of modernisation measures and analyses - related to about 10 issues - should be finalised by the end of next year. Among the already assessed items, no urgent safety weak-points have been revealed.

Among the main improvements, Riskaudit underlines:

- implementation of the LBB concept,
- analytical and experimental demonstration of the bubble condenser functions,
- residual-heat removal by measures on the secondary as well as the primary side, e.g. by the independent emergency steam generator feedwater system, qualified steam dump stations, secondary and primary feed and bleed procedures,
- accident analysis complies to current international practice regarding the scope of PIE’s and the failures analysed and methodology used,
- reliability and qualification of I&C; main and emergency control rooms, ergonomic aspects of man-machine interfaces,
- countermeasures against internal hazards, e.g. systematic fire analysis and fire protection measures,
- seismic reinforcements for equipment,
- operational safety improvements, as implementation of QA program, qualification of the personal at the full-scope simulator and systematic use of operating experience from the Bohunice-3,4 NPP.

In addition to the performed upgrading measures, the good inherent safety capacity of the VVER-440 gives considerable safety margins on all defence-in-depth levels (large volume of primary water, SG water ensures residual-heat removal without feeding for 4-5 hours, stable natural circulation for residual-heat removal up to 9 % of nominal power).

On level 4, the inherent safety capacity is used for effective AM measures. The original emergency operating procedures were event-oriented. For DBAs, symptom-oriented guidelines - so-called Emergency Response Guidelines (ERG) - have been developed together with Westinghouse. They cover not only DBAs but also BDBAs to the level of preventive accident management measures. These guidelines have been verified by using the full-scope simulator at the Mochovce site. The development of severe accident management guidelines for mitigation of severe accidents is planned.

The main remaining issues are:

- probabilistic safety assessment for the upgraded plant status including shutdown events,
- final results of the large-scale Bubble Condenser tests,
- site-related seismic characteristics and eventual relevant upgrading measures,
- completed independent ventilation for main and emergency control rooms (three redundant trains for each one) and PAMS,
- completed protection to high energy pipe breaks on 14,7 m platform.

Furthermore, within the framework of the PHARE project that has been finalised so far, some safety issues which have already been examined by the Slovak side still need further discussion, e.g.:

- completed classification and qualification of the equipment,
- BDBA analysis and corresponding emergency operating procedures,
- updated safety documentation (POSAR) considering commissioning results.

The UJD required safety demonstrations from the NPP in compliance with national regulations and precautionary with respect to the recommendations given by the IAEA and Riskaudit in 1994. Furthermore, for the safety and safety-related equipment delivered from Western countries, the UJD requested certifications according to the procedures valid in the producing country.

It can be concluded that the improved design and operation procedures of the Mochovce NPP, including some few measures which are not yet implemented in the plant, are in accordance with IAEA recommendations.

Basically, they also fulfil Western standards. However some individual deviations from some Western country's practice can be noticed. Nevertheless these deviations are in areas where non-uniformity exists between the Western countries.

On the other hand few Eastern regulatory requirements fulfilled by Mochovce NPP are more stringent than some Western standards, e.g. for reactor pressure vessel integrity (periodicity of reactor pressure tests, in-service inspection and maximum of crack size to be investigated).

At present, at the Mochovce NPP, unit 2, nuclear commissioning is under way. The fuel loading of unit 2 started on October, 4, 1999. The modernisation programme is the same as for unit 1. The current implementation status of the upgrading measures is more advanced than it was before the start-up of unit 1. All measures according the IAEA category III and II already have been realised, except those measures that have not been negotiated successfully with suppliers such as PAMS. Still, the independent ventilation of the main and emergency control rooms will be completed during next year.

For the future operation of both units at the Mochovce NPP, it is recommended to maintain the achieved safety level by a consequent safety-oriented operation,

concerning quality of management, maintenance and operating experience feedback.
Finally, it should be underlined that the Mochovce NPP is the first Soviet-designed NPP finalised in construction in an East-European country which has achieved a safety level comparable to Western standards. The achieved safety status is prospective and guiding for the modernisation of other East-European NPPs of the VVER-440/V-213 type, especially the Slovak units at Bohunice-3,4.

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