# Safety issues for Mochovce 3 & 4 nuclear units

**Assessing the Options to Increase Nuclear Safety at the Planned Completion of Units 3 and 4 at the Mochovce Nuclear Power Plant in Slovakia**

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## 1. Introduction

The Soviet-built nuclear power plant Mochovce in Slovakia consists of VVER 440/213 reactors. They were designed in the 1970i's, at a time when the standard of Soviet nuclear power plants had to be brought up to the standard recognized in the West. The power plants were equipped with safety systems designed to control the rupture of the main cooling pipe of the primary circuit – a standard requirement in new nuclear power plants. The generation of nuclear power plants prior to that– the VVER 440/230 – was not even designed to control such an accident (the first two units of Jaslovské Bohunice, Slovakia, are of this older type) and is together with the "Chernobyl reactors" considered to be the most dangerous in Central and Eastern Europe. Unlike Western reactors in general use at the time, not even the so called second generation VVER 440 was equipped with a containment structure, which serves the dual purpose of protecting the environment from radioactive releases and protecting the NPP from external events (e.g. a plane crash, a shock wave etc). The only exception is NPP Loviisa in Finland, which does feature a containment structure.

**Figure 0a: Cross-cut of VVER 440/213 reactor block.**

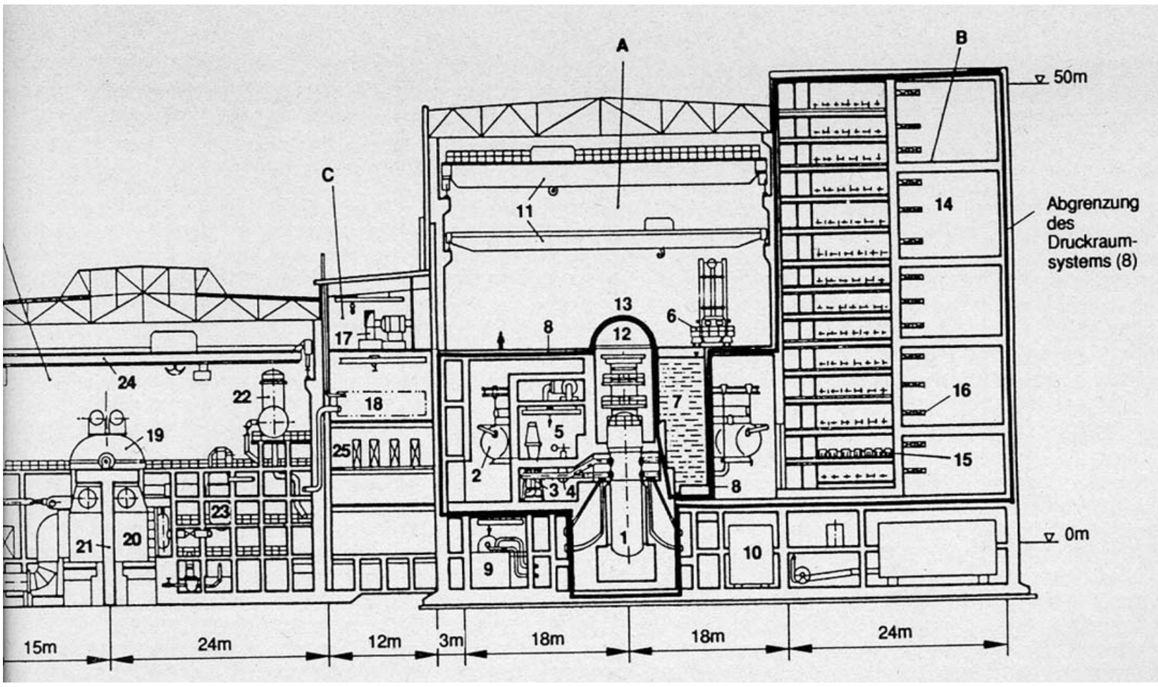
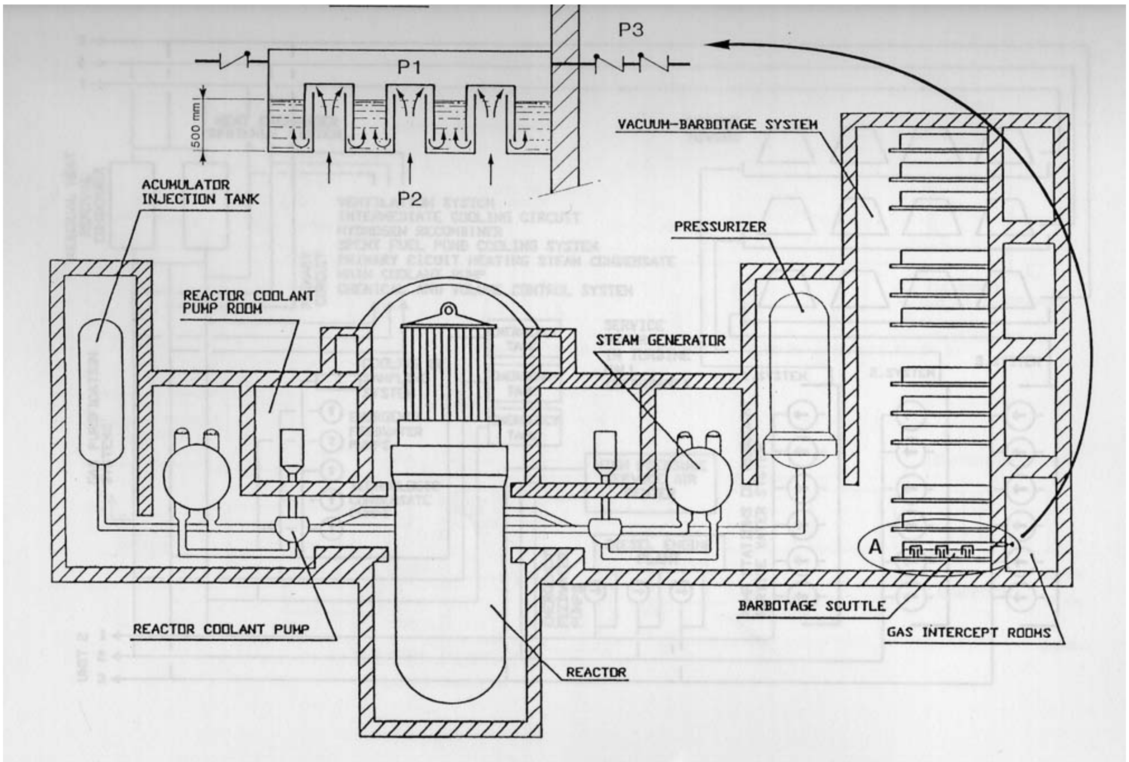


**Figure 0b: Cross-cut of VVER 440/213 reactor block.**




NPP Loviisa confirms that experts were already in the mid-seventies aware of the need to add a containment structure to the VVER 440 reactors, and that the knowledge to implement it was present. It is thus very peculiar that at Mochovce, at the end of the millennium, they built and put into operation two reactors without containment, and what is more is that two additional units are planned for the second millennium. Instead of a containment structure, Mochovce has a so-called bubbler- condenser, which is supposed to reduce pressure and heat of hermetically closed cells and prevent the destruction of the building as well as the release of radioactive substances into the environment after a primary pipe break accident. The function of protection against external events is not fulfilled with a bubble-condenser.

Reactors of the Mochovce type were not built only in Czechoslovakia and the Soviet Union (Rovno 1, 2; Kola 3, 4), but also in Hungary (Paks 1 - 4), in Bulgaria (Kozloduj 3, 4) and former East Germany (Greifswald 5 - 8 – construction and operation were stopped after reunification of Germany for safety reasons). Another plant of this type was partly built in Poland (Zarnoviec 1- 4).

This reactor type was also sold to Finland (Loviisa 1, 2). Due to special economic ties, Finland was obliged to order a nuclear power plant from the Soviet Union. However, the Finnish experts insisted on that key safety equipment would come from western suppliers according to western standards. Thus, these reactors were equipped with a containment structure, with a so-called ice condenser according to US principles, as well as new emergency cooling systems, new Instrumentation & Control Systems from Western Germany (Siemens). The reactors delivered to Cuba (Cienfuegos 1, 2 – with a planned start of operation

1993 and 1996) were also equipped with containment structures. However, construction was stopped due to financial problems and never resumed.

### 1.1. Requirements concerning the severe accident probability of a NPP

Existing risks from operating nuclear power plants, which society is exposed to, can be reduced by improving the safety systems and increasing expenditures for them. This approach, however, is applicable only to a limited extent; in concrete cases the question has to be raised how high the probability for the occurrence of severe accidents is acceptable.

The question of what the highest accepted risk is can be answered with purely economic reasoning, that is by comparing the costs for improving the safety systems to the losses that may occur in case of a severe accident. Even if we ignore the problems connected to such inhumane reasoning, the currently prevailing public opinion fortunately requires a more complex attitude.

The answer to the question of the reactor safety level is that the probability of one single accident with the release of radioactivity into the environment has to be practically impossible. Due to ideas about the development of nuclear energy, where the number of reactors and the overall operation time is increasing, it is necessary to reduce the probability of severe accidents with the release of radioactivity per reactor.

The expected value for the occurrence of a severe accident per reactor per year has to be low enough. This requirement can be formulated e.g. as one accident per reactor per one hundred years, that is one order of magnitude less than the reactor lifetime. The following diagram (Figure No. 1) (value for the occurrence of one severe accident per reactor per year < $10^{-2}$) is based on this requirement (HERMANSKY).

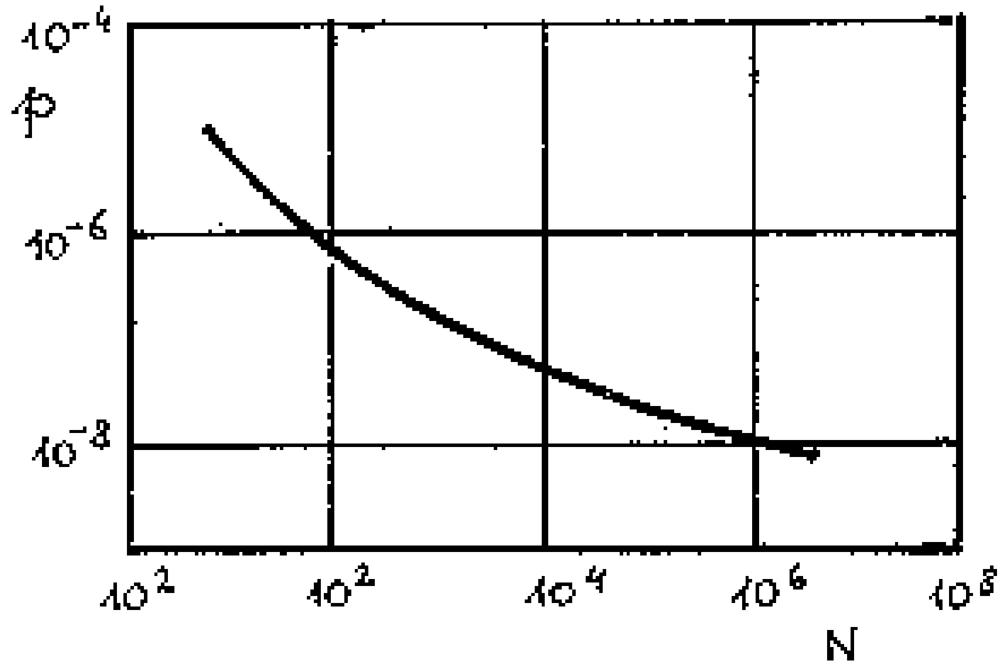


**Figure No. 1: Average maximum permitted probability for severe accidents p in dependence on the overall number of reactor-years N**

Currently operating light water reactors reach a severe accident probability with core melt down of $10^{-3}$ - $10^{-5}$ reactor-year at an amount of reactor-years in the order of $10^4$ reactor-years. If an expansion of nuclear power plants in operation should take place ($N > 10^4$ reactor-years), then the probability should decrease under $10^{-7}$ reactor-years. This clearly shows that in reality the safety level of current reactors should be at least one hundred times higher than it actually is. The higher severe accident probability for the oldest reactors is naturally higher by two orders of magnitude. The frequency of severe accidents during the forty-year history of nuclear energy underlines the accuracy of this consideration.

Therefore it is very important to critically examine the safety level of the reactors at the Mochovce nuclear power plant, especially now when plans to build additional reactors according to designs dating back to the 1970's are under consideration.

## 2. Severe accident probability at the Mochovce NPP

No data on the probability of the occurrence of severe accidents with a fuel meltdown is available for the Mochovce NPP. The CDF (Core Damage Frequency) for the reactors at Mochovce could be estimated by comparing with the Dukovany NPP, for which the results of the necessary calculations are available. The comparison is legitimate because both plants are using the same reactor type.

The diagram on page 2 (MPO) shows that the so-called modernization programme to increase the lifetime of the Dukovany plant started at the value of $1{,}84.10^{-4}$ per reactor-year. At that time construction at units 3 and 4 at Mochovce was standing still. From this we can conclude that it hardly could have reached lower CDF values. The mentioned values can also be seen as a starting point for the following considerations. Should the completion of the two units follow the original concept (EGP), it is likely that the CDF will be lowered. However, it is unlikely to assume that the reduction will substantially exceed the values that Dukovany is supposed to reach after its modernization. Concerning the requirements mentioned in the first chapter, the CDF value will after the modernization be substantially lower than the acceptable nuclear safety level.

**Figure No. 2:**


### 3. Nuclear power plant Mochovce

The Mochovce NPP is situated in the south-western region (Nitranský kraj) of the Slovak Republic, between the towns of Levice and Nitra.
The start-up of the first two units at Mochovce (in the summer of 1998 and end of 1999) represents a strange move in the development of nuclear energy in Europe. This NPP would be hard pressed to get a licence to operate in countries with higher nuclear safety levels.

Chronology of the construction of unit 3 and 4 at Mochovce according to information by the owner (SE):

| | |
|---|---|
| 1980 | Decision to construct the NPP |
| 1987 | Construction permit – start of construction |
| 1992 | End of construction work |
| 1993 | Start of mothballing the plant |
| 2000 | Approval of a strategic plan of mothballing, maintenance and protection |
| 2001 | SE takes over the supplied goods into its ownership |
| 2007 | Feasibility study |
| 2007 | Final Construction / Completion Decision |
| 2008 | Start of completion work |
| 2012 | Start-up of unit 3 |
| 2013 | Start-up of unit 4 |

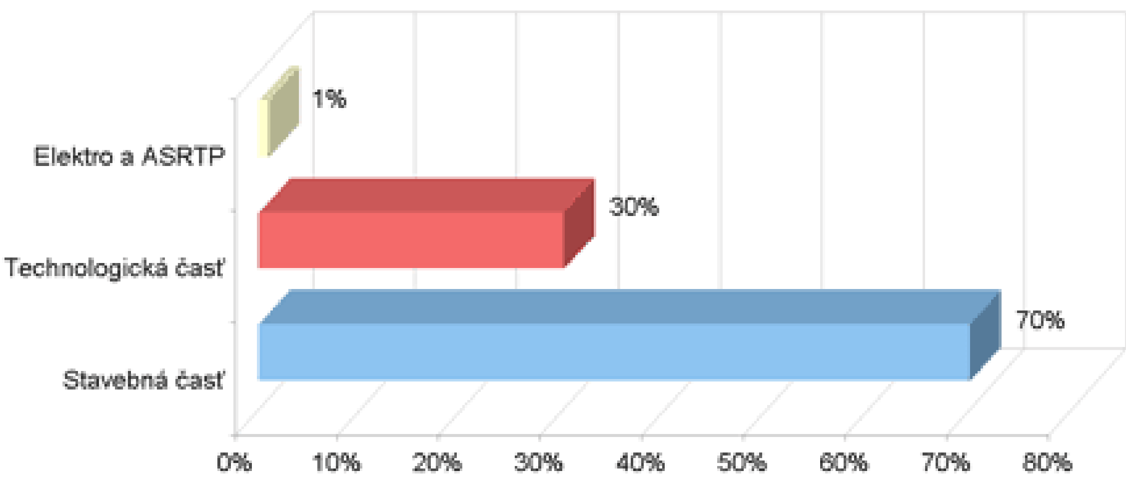
In 1981, when excavation work started at the Mochovce site as a preparation for the nuclear power plant, the plan was to complete all four units. Already in the early 1990's, when the first financial difficulties arose, it became clear that completion would be split into two phases. In the first phase, effort would focus on finishing the first two units.

Construction at units 3 and 4 was stopped right after the completion decision; in 2000 the “Strategic plan of mothballing, maintenance and protection for NPP Mochovce” was set up. The plan contained the following demands:

- Mothballing and protection to maintain the equipment in the quality as demanded by the nuclear authority of Slovakia and recommended by the IAEA
- Preservation of data from this project to secure complete documentation and technical data
- Categorization and specification of individual technological systems and constructions

Units 3 and 4 at Mochovce are directly connected to the currently operating first two units and should make use of some already existing support systems, which are common for all 4 blocs. The completion of 3 and 4 would make use of the second half of the total planned production capacity but with lower financial costs in time compared to the first half. This assumption is based on the fact that all other buildings needed in connection with the construction and operation of the nuclear power plant are already in place.

**Figure No. 3 (SE): Percentage of completion of unit 3 and 4 at Mochovce according to the original plan**




white bar: electric systems and I&C, red bar: technological part, blue bar: buildings and constructions

Data concerning percentage of completion has to been taken with a grain of salt: a certain tendency to overestimate supports the argumentation that the already invested funds are large and costs of completion small. These tactics triumphed when Temelin completion was pushed through, although the data used were utterly unrealistic and constantly changing. There is no reason why the Slovak public should not be subjected to the same demagogy as the Czech public.

And yes: In 1994 (SE&EDF) determined the percentage of completion of unit 3 and 4 at 40% and 30%. On the other hand, if the percentage given would come close to the truth, this would to a very high degree predetermine the design of the reactor which will be built.

In 2006 the so-called SAFETY BOARD MO34 was established. It convenes once a month and is tasked with judging adequacy of the decisions concerning the engineering works for preparation of project 3 and 4 on an international level. Members (SE):

- Maurizio Cumo - Italy – chairman
- Ivo Tripputi - Italy – secretary
- Miroslav Lipár – Slovakia
- Helmut Böck – Austria
- Adolf Birkhofer – Germany
- Leonid A. Bolšov – Russia
- Annick Carnino – France

### 3.1. Nuclear safety problems of VVER 440/213 reactors

### 3.2. Instruments for the assessment of the nuclear safety level Classification of VVER 440/213 safety problems

To assess how severe the individual safety problems of VVER reactors are (a classification also for VVER 1000 is in place), this document (IAEA) is typically used. It defines four categories:

I Deviation from recognized international procedures. It is suitable to include them as a part of activities for the solution of safety issues with higher priority.

II Safety-relevant. Defense in depth is degraded.

III Highly safety-relevant. Defense in depth is insufficient. Immediate corrective interventions are necessary. Provisional measures may be also necessary.

IV The most relevant safety problem. Defense in depth is unacceptable. Immediate intervention is required. Compensation measures must be defined before the solution of the safety problem.

In contrast to the VVER 1000 reactors, where some problems were categorized as IV, (IAEA) did not see any problem serious enough with the VVER 440/213 to label it category IV. Eight problems were classified as category III., 40 as category II and 26 as category I.

### 3.2.1. Problems of category III

#### 3.2.1.1. Qualification of Equipment

The qualification of safety-relevant equipment is necessary to prove their ability to fulfil the requested function. The requirement for qualification concerns the conditions of normal operation, accident conditions and conditions of internal and external events. Experience in nuclear power plants with VVER-440/213 reactors shows that the qualification of equipment is either completely missing or not conclusive.

An example is the qualification of the electrical equipment, the devices and control equipment for accident conditions with loss of coolant (LOCA). The nuclear power plant usually does not have the detailed procedures of the tests nor the original reports about the tests. Moreover, safety reports showed that the originally installed junction boxes of the cables do not necessarily withstand extreme conditions and fail under LOCA accident conditions.

Another example is the lack of qualification of safety-relevant systems (ventilation, pumps of safety-relevant and fire protection systems) under seismic conditions.

<u>Solution for Mochovce units 3 and 4</u>
Suggestions (EGP) concern the qualification of the steam generator safety valves and the steam dump to atmosphere (PSA) functionality. The goal is to prove the qualification for the steam-water and water flow, as well as assuring the robustness of the steam pipes.
In the technological part, the demand is to additionally ground the steam dump valves. For the construction part, a conversion of grounding of the individual PSA screens for the transmission of the new load from the limitation of the vertical shift PSA and the resulting construction changes. At the same time, construction changes may also be the necessary due to improvements of the seismic resistance and pipe whip.

#### 3.2.1.2. Non-destructive testing

Non-destructive tests of the reactor cooling system are being conducted in accordance with the regulations of individual states. These regulations are usually based on the original Soviet codes and standards (with the exception of Finland). Deficits were detected during tests of the reactor pressure vessel from the outside, tests in the area of the vessel cladding, when checking the collectors and tube steam generators, in the limited accessibility of certain welds, reactor lids, reactor lid grommets, pipe welding, welds of the steam generator super-structure and sockets. The non-destructive methods, equipment and personal typically do not deliver sufficiently reliable results.

<u>Solutions for units 3 and 4 Mochovce</u>

The concept (EGP) is based on the modernisation of the primary circuit diagnostic subsystems (especially monitoring of the reactor pressure vessel lid) and adding new subsystems which are needed to assess LBB (Leak Before Break).

It can be expected that the implementation of the LBB concept will encounter the problem of finding out about the history of the components (documentation on production, storing, testing etc.).

**3.2.1.3. Sump net congestion of the core cooling safety system**

During an accident with loss of coolant, the flow of the leaking water or steam can tear down the insulation from the surface of the equipment. Mineral wool with water accumulates on the floor of the respective cell and can congest the water inlet into the sump of the room. This can stop the water supply to the pumps of the safety system in the recirculation phase, which in turn can result in the impossibility to cool the reactor core.

A similar accident happened in Sweden in July 1992. A safety valve was accidentally open during the start-up of the unit and the leaking steam tore down the insulation, which was carried off into the condenser tank. This resulted in the blocking of the reactor core cooling safety system. Tests in Finland and at the Zaporoshe NPP confirmed that in case the insulation is torn off, the sump can be blocked and the pump cannot supply water into the emergency core cooling system.

<u>Solution for Mochovce units 3 and 4</u>
The inlet into the sump at the first two units is equipped with a grid with 10x10 mm squares with a wire of 2 mm diameter. Behind this grid sits a sequence of 7 nets with squares measuring 2x2mm with a wire of 0,63 mm diameter. The super-structures of two sumps out of three are protected by walls, but right above them runs the high energy piping leading out of the hydroaccumulator (constant pressure 6 MPa) and the high pressure injection lines (test pressure 13,5 MPa). The super-structure of the third sump lies exactly between two pumps related to the reactor cooling system. The loop pipes of the main circulation line should be protected against a whip caused by a rupture.

However, in case of a rupture of the above-mentioned pipes, it is realistic that a damage of the sumps due to the flow of leaking fluids and flying debris would occur. The consequences are likely to be similar to the clogging of a sump inlet.

According to the plans for safety measures (EGP), the net construction should be modified on the basis of experimental programmes conducted on testing benches in the Russian Federation and in Tlmače (Slovakia). The programme was designed to examine the behaviour of insulation under LOCA type accident conditions (insulation destruction, drifting, swimming, getting trapped in nets). If the same insulation will be used as in Mochovce units 1 and 2 (mineral wool), a reconstruction of the net is suggested to enlarge the through-flow surface while at the same time keeping the same layout size, which reduces the speed of the fluid flowing through the grid and reduces the clogging of the net. The net walls should be diverted by 20° against the flow of the fluid, which should result in a safe-cleaning mechanism of nets during zero flow through the net. A clogging of the net should be signalled to the control room. Other suggestions include the idea to enable the control system to stop the flow and to perform reliability calculations for the suggested solution.

An alternative is to use another material as insulation (SE&EDF).

The suggested solutions are only directed towards solving the problem of blocking the sump inlet as a consequence of clogging. The problem with the insulation remains. It does not address the problem that the sump is out of use as a result of high energy piping whips in the hermetic compartment and as a consequence of the jet injection of the leaking medium from the ruptured pipe.

#### 3.2.1.4. Reliability of feedwater supply

In nuclear power plants with VVER 440/213 reactors, the heat removal from aftercooling is organized via the secondary circuit under all situations with the exception of LOCA accidents.

After reactor shutdown, the residual heat is removed via the turbine by-pass (steam dump to the condenser) or the technological condenser. If it is not available, then via the steam dump to the atmosphere. The steam generators are supplied via the auxiliary (emergency) feedwater system.

When both feedwater systems fail, the so called super-emergency feedwater delivers water to the steam generators.

In case of a breakdown of the external electricity supply, the aftercooling systems are supplied by the emergency systems – diesel generators and accumulators.

In case of a main steam collector rupture, all quick-acting valves are automatically closed and residual heat removed via the steam dump. However, analyses showed that under certain circumstances during such an accident, a failure could occur that prevents the signal to close the quick-acting valves from being sent.

The super-emergency feedwater system is situated in the generator room. This puts the components of the system (pumps, valves, pipes) into the danger of being damaged by common causes – fire, flooding of the generator room, earthquake. The pipe can moreover be damaged by a steam pipe or feedwater pipe rupture.

The equipment for residual heat removal, especially the electric valve drivers, is not qualified for conditions such as rapidly increasing humidity and temperature, which can occur after a high energy pipe break. The leaked water can penetrate into the lower floors, where the rooms with electric and control systems are situated.

<u>Solution for Mochovce units 3 and 4</u>

The proposed plan (EGP) suggests to prepare a new layout for the seismic piping of the super-emergency feedwater pipes outside the critical space of the +14,7 m level where the high energy pipes are concentrated. The super-emergency feedwater pipe will lead through a new seismic channel into the room under the barbotage scuttle and to the steam generator boxes, where it will be connected to the existing inlet. Not only the layout should be changed, but the EGP also foresees simplifying the piping scheme in the serially connected valves. A new concept aims to also address the independence and the physical separation of the individual power supply systems. Also the super-emergency supply system should be checked concerning potential flooding, fire and seismic resistance, extreme weather conditions and plane crashes.

The question remains, however, to what extent will thorough and consistent implementation of the suggested measures be inhibited by the fact that many of the civil structures are already in place.

However, as the comparison with the previous paragraph listing the problems shows, the solutions cover only parts of a whole array of problems.

#### 3.2.1.5. Bubbler Condenser behaviour under maximum pressure difference under LOCA accident conditions

The mechanical construction of the Bubbler Condenser (walls and coverings) is not satisfactory and there is a risk that the metal structures fail in case of guillotine break of the main circulation pipe.

The efficiency of the system of hermetic cells after such an accident depends on the functioning of the Bubbler Condenser system. If the Bubbler Condenser system fails in the initial phase of the accident, the water from the Bubbler Condenser's channels can flow over into the Bubbler Condenser tower. This can also cause sudden steam condensation and a pressure reduction in the hermetic rooms, which results in the loss of water in the Bubbler Condenser's channels. If the failure occurs in a later phase, the Bubbler Condenser will not be able to fulfil its function according to the design.

Calculations discussed in the framework of the IAEA showed that a number of features of the Bubbler Condenser system need to be strengthened (carrier beams, strengthening of wall ribs).

Analyses were conducted to test the design pressures and temperatures concerning the design basis accident (break of the main coolant pipe 500 dy) under several accompanying conditions (GRS).

The results of these calculations showed that, depending on the accompanying conditions, the design basis parameters for pressure and temperature of the hermetic zone (245 kPa, 127°C) were reached, and possibly slightly exceeded.

Analyses pointed out that the integrity of the cover during the process when the steam-water mix bubbles through from the hermetic zone (more than 2 covers) the design pressure will be exceeded. The failure of 12 covers in one channel already results into pressure as high as the pressure value which generates if no water would be in any of the channels of the whole Bubbler Condenser system. Therefore it is necessary to reassess the strength of the covers for different dynamic loads.

Analyses pointed out, that only very limited, sometimes no safety margins are available and that the failure of only a few covers in the channels already leads to exceeding the design pressure and temperature.

The sizing of the hermetic zone was based on the assumption that 30 minutes after the main coolant pipe break a renewal of under pressure in the hermetic zone occurs as a consequence of steam-air mix has condensed. This assumption could not be confirmed yet, since up to now it was not possible to provide sufficient calculations on the long term behaviour of the leak from the ruptured pipe.

The fact that the Bubbler Condenser system was never tested in a test facility (1:1), is being criticized in all safety analyses for plants with VVER-440/213 reactors.

<u>Solutions for units 3 and 4 Mochovce</u>
Surprisingly, this problem is not even mentioned in the plan for improving nuclear safety. To improve the Bubbler Condenser System is not even considered, not even in the maximum version (EGP).

### 3.2.1.6. Fire prevention

Safety reports about nuclear power plants with VVER reactors identify a number of weak points regarding fire prevention, which in many cases differ from the safety recommendations of the document IAEA NUSS Safety Guide 50-SG-D2. Redundant systems, components and cable raceways of safety-relevant systems are in some sections located without sufficient physical separation and not protected against the spread of fire. This is apparent by the following examples:

- lack of antifire doors and barriers with corresponding qualification
- lack of antifire sprinklers in the ventilation channels
- redundant cable raceways are positioned closely next to each other
- lack of qualification of the penetration coating
- cable coatings are not fire resistant

A fire could therefore result in the failure of more than one redundancy of the safety-relevant system. This could be e.g. important parts of the system needed to remove residual heat from the reactor, like the feedwater pumps, emergency cooling water pumps, separation and safety valves of the main steam pipes. In addition, the fact that the valves on the feedwater line are located in the open space in the machine hall and shared by both units, could make them vulnerable to a fire of the turbine oil system.

The shared machine hall for two units is a very unfortunate solution especially for fire safety and the possibility of impacting the equipment of the respective unit, where the abnormal situation took place. Due to the high fire potential in the turbine area (e.g. 25 $m^3$ turbine oil with a flash-point of 180° C in the main oil tank for one single turbine, that is 100 $m^3$ in the whole machine hall, wire coating etc.) and the high number of potential flammable points cannot exclude an extensive fire. Because the machine hall contains safety-relevant equipment, a big fire can lead to the failure of several pieces of equipment, which are not separated from each other from a fire protection aspect.

In general, the VVER-440/213 does not have strict separation between the cable lines of the redundant systems. In some parts the power cables and control system cables of redundant components are situated in the same fire separation zones. In such cases a fire could have safety consequences caused by a common cause accident. The cable rooms under the control rooms and the emergency control room contain a high amount of cables connecting the control rooms with the safety systems, which penetrate the ceilings of these control rooms. The cable segregation of the respective safety trains is not sufficient. This is a serious problem, since a fire in one of the rooms can lead to loss of control over all three safety systems. It is not clear whether a fire among the control room cables could also disable the emergency control room.

Another problem concerns the insufficient protection against the ignition of oil. The pieces of equipment filled with oil do not always have a storage reservoir to catch leaking oil. The flanged joint of the oil pipes do not have sealing pads and casings. The shutdown valves on the ventilation pipes leading into the rooms with the oil tanks are designed without protection against sparks. In those rooms, the heating devices are not being monitored and the fire doors are not designed to withstand the pressure caused by an oil explosion.
A specific concern regarding oil burning is the oil greasing of the main cooling pumps. At the blocs with VVER-440/213 reactors the main and the auxiliary circulation pumps have their own oil system. The rooms with the oil tanks and oil pumps and the room with the main circulation pumps engine are not protected in the original plan of the project. The possibility of an oil leak and the presence of components with high temperature create a high risk that a fire might break out.

<u>Solution for Mochovce units 3 and 4</u>
The issue of fire protection is not yet finalized. It will be reviewed on the basis of an analysis of the situation at unit 1 and 2. The review for unit 3 and 4 should contain the differences in solving the fire prevention issue based on the difference between the two dual-blocs, especially under the aspect of solving the cable distribution system.

It is necessary to assess whether the requirements and implementation of measures against the spreading of fire between fire compartments inside the secondary circuit and between the secondary circuits of both units comply. The task is mainly about the following issues: making sure that the fire resistance of the load-bearing and separating fire protection constructions is sufficient; fire doors and whether they are smoke proof; fire dampers; tightening of the grommets; fire insulation of ventilation channels; fire sealing of construction joints inside the secondary circuit etc. Fire dampers remotely controlled by the electric fire signalisation, seismically resistant, have to be used.

From a fire protection aspect, the control room and the emergency control room have to separate.

The need for local separation of the cables, measures to separate redundancies, and protection of the connection between the control room and the emergency control room still has to be assessed.

Another important measure is to reduce the risk of an oil leak and the complete enclosing (fire compartment) of oil tank of each turbine with a lid and a storage reservoir.
The central oil system has to separate in the sense of fire protection.

The measures on the 14,7 m level are to be reviewed based on the assessment of the impact of fire on the equipment installed on this floor.

It is necessary to update the principles for separating cable raceways belonging to the respective safety systems.

The fire protection seems still to be under development. Comparing the data now available it is possible to conclude that the IAEA recommendations will not be implemented completely.

#### 3.2.1.7. Risk of high energy pipe break

Calculations showed that the break of a high energy pipe (pressure > 2 MPa and temperature 100°C) can initiate the following dynamic effects:

- pipe whip as a consequence of the reactive forces
- clash of a flow of steam or fluids shooting out of the broken pipe.

A pipe whip or gush as a consequence of a high energy pipe break should not cause the worsening of the initial event (which is the pipe break) or damage safety-relevant equipment responsible for getting the initial failure under control. The dynamic accident effects should not disable the possibility of a reactor shutdown and keep it in the status after the initial accident. At risk is however the secondary circuit. The zone between machine hall and the reactor hall is vulnerable due to the presence of many pieces of equipment, such the main steam collector, the feedwater pipe, the emergency feedwater pipe, etc. on the 14,7 m level and below.

This leads to the risk of a multiple failure of safety-relevant systems. Protection of reactor pressure vessel integrity and loss of feedwater belong to the most serious problems of a nuclear power plant with VVER reactors.

To prevent the reactor vessel from cooling down too much and the consequences to the reactor pressure vessel as well as excluding the possibility of recriticality, it is necessary to examine an accident with the main steam collector rupture and identify the number of steam generators, which can be affected and will be supplied by the ruptured pipe part. It is also necessary to include the cases where the pipe whip causes damage to other pipes or valves on them. Damage to the emergency feedwater system can even lead to loss of coolant on the secondary site.

Solution for Mochovce units 3 and 4
Measures against pipe whip (EGP) of the broken pipes have been taken into account. The critical pipe spots should be identified, using the method which was already applied for the first dual-bloc. Implementation of a solution should be easier to undertake since the technology has not yet been installed on the second dual-block.

Exactly for this reason we have to raise the question: why it is not possible to apply the practice from countries with a developed nuclear safety culture and provide physical and structural separation between vulnerable sections? Pipe whip restraints can reduce or prevent a pipe whip. However, it does not solve the consequences of the clash of leaked medium fluids coming out of the ruptured pipe.

### 3.2.1.8. Seismic measures

The seismic measures as well as the input values for the analyses in general do not correspond to the international standards. The resulting structures, components and distribution system do not have enough safety margins necessary for the design basis earthquake (IAEA).

Solution for Mochovce units 3 and 4
The plan for the structural details for the seismic improvements (type of partition walls, grounding of circumferential cases) should take the current stage of construction work into consideration (EGP).

The specification of the amount of work and the cost estimate are based on the assumption that the existing parameters of the seismic design basis for the site NPP Mochovce, i.e. keeping the maximum horizontal ground acceleration 0,1 g and assessing the frequency content with the NUREG spectrum. Any increase above the value of 0,1 g would mean a distinct difference concerning the strengthening and an increase in costs in the construction and in the technological parts of the plant (EGP). In this category would be the plan of rerouting the pipe on the +14,7m level and the steam pipes in the steam generator boxes.

The seismic measures seem to be the issue where the stage of completion determines the safety level. Each and every concrete seismic measure therefore has to be assessed while keeping in mind that there might be the attempt to lower construction costs at the expense of nuclear safety.

### 3.2.2. Category II problems

### 3.2.2.1. Component classification

The components for nuclear power plants with VVER-440/213 reactors were designed on the basis of the Soviet safety regulation OPB-73. This regulation discusses five classes of safety-relevant components. In July 1990, regulation OPB-88 came into force defining a new classification. The components of one class required that the same quality level has to be kept up during design and production. Re-classifying of the components led to non-uniformity in the quality assurance requirements.

Solution for Mochovce units 3 and 4
(EGP) suggests to prepare an inventory of selected pieces of equipment, to discuss them with Nuclear Regulator with the goal of reducing the scope, (!) and to compile a component classification concerning nuclear safety and seismic resistance.

#### 3.2.2.2. Reliability analyses for the systems in Safety Class 1 to 2

Reliability analyses for the systems in Safety Class 1 to 2 are indispensable to be able to confirm that the systems are as reliable as the designer assumed that they would be. It is also important to gather data about component reliability during operation, and to confirm the validity of the original analyses. Some power plants do not even have complete analyses for the construction phase, nor do they systematically collect data during operation.

Solution for Mochovce units 3 and 4
(EGP) promised to prepare a reliability analysis with the help of the fault tree and taking uncertainties and risk factors into consideration.

#### 3.2.2.3. Preventing unintended dilution of boric acid

The main reasons for a transient to occur as a consequence of a boric acid dilution in the cooling water are:

- when a primary circuit loop is reconnected after shutdown to the system
- dilution caused by the chemical system and the pressurizer system, especially after main circulation pump failure following a power loss,
- as a consequence of a leak in the heat exchanger system of the emergency core cooling system.

Unintended dilution is usually detected by measuring the neutron flux. With this method, the operator’s intervention time is relatively short and it is safer to detect the dilution earlier by measuring the boron content. To use this method effectively, the time of response has to be short (around 10 minutes) and the precision sufficient (not more than 100 ppm).

A continuous monitoring of the boric acid concentration is conducted with a boron measurer, which is not sufficiently precise and has a time lag between the current and the indicated value. Neutron flux monitoring during reactor outage, or in the phase of reactor start-up, is difficult since the rate in the ionising chambers is very low ($10^4 – 10^3$ n/cm$^2$.s) and cannot be measured with the currently existing pieces of equipment (AKNP-3). The monitoring relies on acoustic signalling. For the reasons mentioned above, usually a neutron source in the core is used to increase the neutron flux during reactor outage or during start-up (international best practice).

Solution for Mochovce units 3 and 4
(EGP) intends to analyse in detail the possibility of non-intentional dilution of boric acid in the primary circuit and to prepare possible scenarios which would completely cover the identified spectrum. Based on IAEA recommendations, acceptability criteria should be formulated in relation to the probability for the event to occur. Moreover, a detailed probability analysis for the frequency of selected events will be performed. Based on the analysis, improvement will be suggested.

### 3.2.2.4. Pressure vessel integrity

The reactor pressure vessel of the VVER-440/213 reactors is exposed to fairly high neutron flux. Mainly the circumferential weld opposite the core is affected. Witness samples are used to monitor the embrittlement of vessel material. Samples are placed in casings which are installed at the external surface of the core basket. The casings are arranged in six double chains, two form a set. The samples represent the basic material of the vessel, the weld, and the material under radiation influence. In the casings, the flow is being monitored (foil made of iron, copper, niobium and cobalt), as well as the temperature (diamond powder).

The most serious problem for reactor pressure vessel integrity is pressurized thermal shock, which can be caused by the activation of the emergency core cooling system or by heat removal in the secondary circuit. To protect pressure vessel integrity it is necessary to implement measures to reduce stress caused by a pressurized thermal shock. The water in the tanks of the emergency core cooling system should not be heated up according to original design. However, analysis showed that cold water injection into the reactor vessel after it showed signs of embrittlement can lead to an unacceptable risk. Therefore, the recommendation to heat the tanks of the emergency core cooling systems was issued.

Solution for Mochovce units 3 and 4
(EGP) considers the standard programme of witness sample monitoring to be insufficient. They suggest the implementation of a modified programme to eliminate these deficiencies. The programme foresees the following measures: collect information about the neutron flux rate with witness samples inside the pressure vessel with 20% accuracy; introduce the measuring of irradiation temperature with the help of fusion monitors and thermal elements; a guaranteed fixed position of the samples toward the core; make an assessment of the degradation processes due to radiation possible with direct methods; and deliver reliable data about the radiation damage of the reactor pressure vessel material during the lifetime of the bloc.

The protection against pressurized thermal shock is not addressed. It is legitimate to assume that the tanks of the emergency core cooling system will be heated up, as is the practice at the first dual-unit (SE-EDF).

### 3.2.2.5. Main coolant pipe whip

The main coolant pipe is made of austenitic stainless steel and welded with many different welds. The pipe whip restraints were found to be incomplete in many plants – distance pieces were missing, which should have been there according to the design. Such pipe whip restraints cannot fulfil the requested function.

<u>Solution for Mochovce units 3 and 4</u>

The LBB (Leak Before Break) concept is suggested to replace the incomplete pipe whip restraints. To insure a reliable detection of a primary circuit leak, three independent leak monitoring systems are to be installed.

As was already pointed out in chapter 3.2.1.2, the implementation of the LBB concept encounters the problem of incomplete records of the individual components (documentation regarding production, storing, and testing etc.). (EGP), however, plans to base the status LBB on a system of leak detection and the results of preoperational and operational controls. However, it has to be understood that leak detection as such already assumes that the LBB requirements are being fulfilled.

### 3.2.2.6. Primary steam generator collector integrity

Every VVER 440 reactor has 6 steam generators which contain two cylindrical collectors (hot and cold leg) and create the border between the primary and the secondary circuits. Heat-exchanging tubes are connected to the collectors, which are made of austenitic stainless steel. The main shutdown valves are installed on the primary circuit pipe and will separate the steam generator in case of an accident. With the exception of NPP Loviisa however, those shutdown valves are not classified as safety-relevant equipment.

In some currently operating power plants, corrosion cracking in the area of the welds on the primary circuit have been observed, most likely caused by the coolant chemical system. It was found that the collector cover is being lifted during operation and this is caused by corrosion cracking in the connecting joint. A potential cause for those effects can be an inadequate chemical regime, or an inadequate chemical composition of the grease used for the screws and inadequate maintenance procedures.

<u>Solution for Mochovce units 3 and 4</u>

(EGP) intends to conduct probability analyses on the steam generator breakaway of the primary collector lid in several scenarios, calculations on strength and lifetime of demountable collector joints, and calculations on the residual stress after blanking the leaking pipe. Documentation on the impact of the planned measures on the lifetime of the threads ,measures to improve the quality of the maintenance procedures and to assess the impact on the safety and residual steam generator lifetime.

### 3.2.2.7. Steam generator tubes integrity

Degraded steam generator tubes with cracking going through a certain wall thickness are usually blanked or equipped with a thread according to the criteria which were developed using the stream analysis results. The original method assess the steam generator state monitors the bubbles inside the collector with a camera, while the secondary side is drained and put under pressure with gas. The activity of the secondary circuit is also measured. This method does not detect a tube unless the crack in the tube goes through the whole wall. Nor does this method monitor and predict the defect behaviour during an inspection. Testing with the eddy current will be introduced. The technically substantiated criterion for blanking the tubes and the limits for leaks are not introduced in all power plants.

Solution for Mochovce units 3 and 4
(EGP) builds on the measures already adopted at the first dual–bloc where analyses on the probability of a break of the steam generator heat exchanging tubes were undertaken. Calculations were also made regarding the tube's elastoplastic deformation in the case of the occurrence of through-holes. Characteristics for the slow spreading of corrosion cracks und breaking characteristics for tubes were identified. The available documents do not mention which measures were taken to address the issue.

### 3.2.2.8. Primary circuit protection against cold pressurizing

During cooling down or during reactor outage there is the possibility that the primary circuit will be cold pressurized. This can have negative effects on the pressure vessel integrity.
Causes for cold pressurizing can be:

- false activation of the emergency core cooling systems
- accidental opening of the safety valves, start-up of the emergency core cooling systems and filling up the reactor vessel with cold water, shut down of the safety valves and the pressure increase at low temperature of the reactor cooling system
- a mistake by operator during hydrotests

In case of a leak in the reactor cooling system, the accident procedures prescribe to localise the leak as quickly as possible. However, the reactor is not protected against cold pressurizing the primary circuit, which probably can occur in the course of the accident. Studies examining the risk of pressurized thermal shock indicate this risk.
Plants with VVER 440 reactors have implemented measures to prevent the cold pressurizing of the primary circuit. However, those measures consist at best of administrative procedure and of human intervention. In Western plants, the protection does not only rely on the action taken by the operator, but also on the activation of the automatic valves.

Solution for Mochovce units 3 and 4
(EGP) intends to prepare and introduce regulations aimed at ensuring that the initial event (that leads to transients during cold pressurizing) is removed. (EGP) further intends to define and introduce an automatic control system, which will be able to prevent thermal shocks during transients and to guarantee pressure vessel integrity. The solution will consist of an automatic shutdown of equipment that could cause increased pressure in the primary circuit up to a value close to brittle break of the reactor pressure vessel during start-up or shut-down of the

unit. At issue are the high pressure pumps of the emergency cooling system, auxiliary pumps, the high pressure pumps of the boric concentrate, and the electric heater of the pressurizer. This solution also includes an automatic opening of the pressurizer relief, possibly even the safety valves based on monitoring pressure and temperature of the primary circuit.

#### 3.2.2.9. Mitigation of the consequences of the steam generator primary collector break

A unique design of the steam generators in nuclear power plants with VVER-440 is the cylindrical primary collector with a screwed flange. A break here cannot be excluded (e.g, at NPP Rovno, the lid opened in 1982). Possible initiating events can be:

- Oscillation of the water surface and a local surge in such a way that the water surface level reaches up into the area of the primary collector lid, followed by dirt accumulation on the screws and the resulting corrosion. This effect can mainly be observed close to the hot collector.

- Deterioration of the screw threads as a consequence of frequent maintenance work. At NPP Rovno the screws were very tight to prevent leakage via the collector lid seals.

- Hydraulic shock during the transient initiated by the pressure loss in the reactor cooling system. The steam leaking from the core flows and collects in the case in the upper part of the steam generator collector. A possible water injection from the emergency core cooling system and the successful isolation of the leak leads to: a pressure increase in the emergency core cooling system; steam condensation; and a hydraulic shock in the collector head of the steam generator.

A lifting of the lid causes a leak from the primary circuit equivalent to a 100 mm diameter (to the secondary circuit). This leak can cause a by-passing of the hermetic box systems with the bubbler condenser systems since the primary circuit coolant leaks directly into the environment, that is, if the safety valves open and the attempt to close them fails. During the transient, even the water supplied by the emergency core cooling system is lost instead of being pumped back from the floor of the hermetic box and returned into the circuit. Moreover, the water injection from the ECCS tends to keep pressure in the reactor cooling system. Because the steam generator safety valves are not qualified for the steam-water mix flow, they can fail during repeated closure. In this case, the direct release path for the primary coolant is directed into the atmosphere. Such a release includes also radioactive substances from the fuel, which can be set free from the fuel during this transient. In the original project design, no such barrier was considered, and therefore no preventive measures are identified.

<u>Solution for Mochovce units 3 and 4</u>
(EGP) suggests adding auxiliary water injection trains into the pressurizer from the pumps with a regulating and closing valve. Moreover, a measuring system for radioactivity detection on the steam generator second side is planned.

### 3.2.2.10. Main coolant pump packing cooling system

During normal operation, water circulates through the packing cooling system of the main cooling pump in an autonomous circuit. More packing water comes from the reversible pump.

When a power loss occurs, in some nuclear power plants, even the reversible pumps stop because they are not supplied from another power source (diesel generator). In this situation, the packing cooling system of the main coolant pump guarantees the water circulation in the primary circuit to cool the components (one circulation pump belongs to each main coolant pump, this autonomous injection circuit prevents the hot water from reaching the main coolant pump bearing). When one of those circulation pumps fails, the packing of the respective main coolant pump is not cooled. This initiating event can lead to primary coolant loss via the seals if it is not being cooled for a longer time. Via the damaged seal, water escapes into the room with the main coolant pump engine outside the hermetical zone and thereafter disappears for a possible recirculation phase of the emergency core cooling.

Experiments showed that more than 50 litres of water can leak in 24 hours via the seal.

<u>Solution for Mochovce units 3 and 4</u>
(EGP) plans to regulate the system of controlled leaks from the main circulation pump in such a manner that the drainage of the main coolant pump board is lead into the steam generator box and collected there to be used for the ECCS.

### 3.2.2.11. Qualification of pressurizer relief and safety valves for water flow

Under certain accident conditions, the pressurizer valves can work with the steam water mix. In case of a failure of re-closure, a small accident of the LOCA type can occur.

<u>Solution for Mochovce units 3 and 4</u>
The goal is to achieve the qualification for the long-term operation during the flow of the steam-water-mix. This is to be achieved with the following measures: replacement of the solenoid valves with valves that have electric drivers; adding a new pilot valve for the relief valves; and a diameter increase of the relief valve pipe. Another measure within the framework of this solution is the change of the disposition and dimension of the blow-off pipe of the pressurizer safety valves (PSV), as well as supplementing the blow-off of the incoming pipe PSV and addition of a nitrogen supply into the PSV exhaust (EGP).

### 3.2.2.12. Integrity of suction pipe systems of the ECCS

The sumps of the system are situated in the reactor hall on the lower floor of the hermetic zone. The suction pipe penetrates the floor of the hermetic zone and leads into the room with the boric water tanks located 6 m below. Then the suction pipe extends through the floor of the room and leads to the heat exchangers, which are placed 12 m below the floor of the hermetic zone. This part of each route is equipped with a separating valve.

When a suction pipe breaks, water can get lost from the hermetic zone. The consequence is a ECCS system failure. This can uncover the core and lead to a serious accident.

### 3.2.2.13. Heat exchanger integrity in the ECCS

Under LOCA accident conditions, the residual heat is removed with the heat exchange of the low-pressure emergency regeneration system. They are cooled with the important service water, which transports the heat directly into the sprinkler tanks or the cooling towers. There is no closed cooling inter-circle. Because the heat exchangers are continuously supplied with raw water, they risk pollution and a degraded cooling. This can increase the risk of leaks in the heat exchanger. When the leak leaves the ECCS, radioactive substances can reach the environment.

An unobserved escape from the important service water system into the emergency core cooling system in the heat exchanger can cause a dilution of the boric acid and possibly create a clean water pocket on the side of the emergency core cooling system in the heat exchanger. At the beginning of the recirculation phase, this clean water could reach the active zone and carry with it a strong positive reactivity.

Solution for Mochovce units 3 and 4
**(**EGP**)** suggests preparing a solution with a system with water circulation in the intertube space via a measuring sensor, which would in case of a leaking heat exchanger signal the change in the medium conductibility. After laboratory confirmation of the results from the samples taken and that service water did penetrate in the boric acid, the operator would shutdown and cool down the bloc and the heat exchanger would be repaired.

### 3.2.2.14. Qualification of steam generator relief and safety valves for water flow

The relief valve-steam dump to atmosphere (PSA) and two safety valves are installed before the main quick acting valve. In case of a leak from the primary into the secondary circuit, the water from the first circuit can quickly fill the steam generator and the steam pipe up to the PSA, which cannot be separated. The lack of qualification of the relief and safety valves for water or water-steam mix can lead to failure in open position. This would be a by-pass of the hermetic zone with a leak of radioactive substances from the primary circuit. Long-term cooling can be put at risk due to loss of coolant into the atmosphere.

Solution for Mochovce units 3 and 4
The goal of the plans (EGP) is to prove the qualification for the flow of the steam-water mix and water including verifying strength resistance of the steam water pipe.

For the technological part, the demand to additionally ground the sump to the atmosphere valve (PSA) should be considered. For the construction part, a conversion of grounding of the individual PSA screens for the transmission of the new load from the limitation of the vertical shift PSA and the resulting construction changes. At the same time, construction changes may also be the necessary due to improvements of the seismic resistance and pipe whip.

#### 3.2.2.15. Functioning of steam generator relief and safety valves under low pressure conditions

At most power plants the current version of relief and safety valves does not allow residual heat removal via steam discharge into the atmosphere under low pressures (lower than 3 MPa) if the steam pipe separating valves are closed. The safety valves open and close at pressures around 5,7 MPa and 4,7 MPa. They function based on a system of compressed air. The opening pressure can be reduced to a value of around 3 MPa, but not less. The relief valves are engine driven and can open under any pressure. However, in the majority of plants, those valves are installed on the main steam line behind the separating valves, which means that they are separated from the steam generator under accident conditions. This solution limits the cool down potential.

<u>Solution for Mochovce units 3 and 4</u>
The plan (EGP) does not treat this issue.

#### 3.2.2.16. Control room ventilation

The original control room design does not suggest a ventilation system able to filter the incoming air in case of radioactive contamination. The risk of inhaling contaminated air in the control room is high. The inhabitability of the control room has to be guaranteed also in case of an accident with serious consequences.

<u>Solution for Mochovce units 3 and 4</u>
(EGP) suggests a completely new ventilation system for the control room.

#### 3.2.2.17. Hydrogen removal system

After some time following a LOCA type accident, hydrogen develops due to a chemical reaction of metal with water and radiolysis in the core. Under the design basis accident conditions, this is a long term process, which, depending on the produced amount, leads sooner or later (over the course of several weeks) to a concentration of hydrogen exceeding the ignition point (> 4%). In case of severe accidents, hydrogen accumulation can occur much faster. One of the main causes of hydrogen leaks is the reaction of water with the aluminium cover of the primary component insulation. The inhomogeneous distribution can cause locally higher concentrations, which reach the ignition point earlier than with homogenous distribution. The original plant design does not include measures to contain this effect inside the hermetic zone.

<u>Solution for Mochovce units 3 and 4</u>

According to (EGP), a monitoring and evaluation system with 8 sensors and 16 passive catalytic recombinators should be used. To solve long term post-accident processes, the addition of lighters to the system (sparkers) is under consideration.

### 3.2.2.18. Primary circuit ventilation under accident conditions

One of the important findings from the 1979 Three Miles Island accident was that during transient processed with pressure decrease, non-condensable gases and hydrogen can be set free. When in addition water spills from the ECCS accumulators into the reactors under low pressure, nitrogen is set free, with which the water in the accumulators is saturated. These gases then accumulate at the highest point of the primary circuit (the highest lying spot of the primary circuit loops is the space under the reactor lid). Design basis accident analyses assume that the globular openings of the hydro-accumulators can prevent the nitrogen from entering the primary circuit after emptying the accumulator (a nitrogen pillow can float above the water surface). If this assumption is not valid, a significant amount of non-condensable gases can penetrate into the upper part of the reactor vessel. The gas can block the primary coolant circulation and lead to overheating of the core. A problem can occur during a small LOCA, e.g. after a leak on the pressurizer steam side. One solution to the problem could be the installation of a ventilation system for the space under the reactor lid and other high-lying points in the primary circuit.

<u>Solution for Mochovce units 3 and 4</u>
The plans (EGP) do not look into this issue.

### 3.2.2.19. Important service water system

All safety-relevant equipment and components are cooled with so-called important service water. In spite of its high importance for safety, it is not clear whether the system is protected in all power plants against all sources of common cause failure including external risk. Redundant important service water trains are without separation or are close to high energy pipes.

<u>Solution for Mochovce units 3 and 4</u>
(EGP) does deal with the service system, but not, however, with the issues that the IAEA criticized. It solves a potential system failure, choking up the exchangers, their corrosion and temperature regulation in the ventilation tower pools and suggests a protection of the built-in-system of those towers under extremely cold weather conditions.

### 3.2.2.20. Instrumentation and Control System (I&C)

The reliability of the I&C consists of two aspects:

1) Accessibility of the system
   The I&C system used in plants with VVER-440/213 stem – concerning their technical level - from the early 1970’s in the Soviet Union and their

reliability is under dispute. The points of criticism are bad quality, aging, complicated maintenance and the need of frequent care.

2) Dimensioning of the I&C
   The resistance against simple failure is not fulfilled everywhere and in some systems, there is no adequate physical separation of the redundant components.
   The accident protection panels consist of two independent trains, which are located in one room. This can lead to the complete loss of the system if a common mode failure occurs, e.g. as a consequence of a fire.

In some sections, the cables of all systems are located close together, e.g. under the control room, which is risky if water penetrates into the room. The water can penetrate to the cables not only if a water pipe breaks, but also if the air conditioning leaks, which already has happened in one nuclear power plant. A water leakage or fire can cause a complete loss of the reactor protection system.

Because the cabling in the original design was not fire resistant, in some plants the fire resistant cover was added later. This of course increased the weight of the cables on the cable trays and basically prevented any exchange of the old cables against new ones.

<u>Solution for Mochovce units 3 and 4</u>
The I&C should be solved anew (EGP). The concept is fundamentally different from the concept applied in the first dual-bloc. The system should integrate on a higher level the function of protection, control and information. It is to include on the one hand the subsystem of reactor protection and control and the measuring inside the reactor, and on the other hand the control of the safety systems and the systems related to nuclear safety. The systems for the control of normal operations should include information and operational control at stable and defined transients as well as the regulation of important values of the bloc and the technological component protection.

The plan is to install a modern modular multilevel bus system, which should as much as possible use computer technology. The safety functions´ and interventions´ back-up should be relying more on ´classic´ control connected to the electronic equipment of the plant. It should have auto-diagnostic functions to a much higher degree than the equipment and systems have in the originally planned project. This should increase reliability and availability of the safety control, as well as the control systems for normal operation.

For some sensors a shared use is planned. This will also result in a uniformity of set-ups of activation of systems which serve to back up each other and to a reduction of costs for purchase, calibration and maintenance. The question remains, however, whether this on the other hand does not reduce the resistance against common mode failure?
The application of the new system also means that there has to be a new solution for the control room.

### 3.2.2.21. Overview of signals initiating a reactor scram

The safety analyses examined the effects of the emergency protection signals of the type HO-2 or HO-3, which initiate an output limitation or its reduction by inserting control assemblies into the core. If HO-2 or HO-3 fail, it is necessary to generate a signal to activate protection HO-1.

<u>Solution for Mochovce units 3 and 4</u>
According to (EGP), the first-cause signalling of HO-1 should be extended by including high temperature in the hot leg of the loops, high water level in the pressurizer and high pressure of the primary circuit coolant.

In the framework of the modernization of the reactor protection system a physical and functional separation should also be undertaken. At the same time should be solved the question of the auto-diagnostic system including the sensors and information output for the personnel at the control and the emergency control room and into the information system.

### 3.2.2.22. Ergonomics of the control rooms

The control rooms of plants with VVER-440 reactors enable the control and indication necessary for the operator to perform the actions required during normal operations and shut-down of the reactor. It is a classical break-up into subsystems with ´blind´ schemes of each subsystem and pilot-lights of pumps and valves signalling their respective functional position. This type of organization led to operational problems, most famously at Three Miles Island, because the operator’s attention is focused on one specific thing with the tendency of inattentiveness to the interaction of the systems.

However, there are other deficiencies in the ergonomics in comparison to modern standard. The indicators for measuring the different types of operational values, e.g. flow and pressure, are not discernible without the help of the legend. Switches for the pumps, the valves etc. have the same form of controller. The indicators that mediate data important for the operator to evaluate the safety status of the blocs are not differentiated from the indicators used for normal operations. A significant part of the valuable space on the panels (directly opposite the operator’s view) is taken up by indicators for rarely used activities related to the reactor start-up and control testing.

According to the original project, the signals are divided into two groups – emergency and warning. Those groups are differentiated by their placement. They are also differentiated by colour and sound. Besides this, the first signal of a certain activity is key for the reactor shut-down system. However, the amount of signals in the control room is too high and the prioritisation is not sufficient.

The setup of the information displays in the control room does not give the operator a quick overview of the information concerning the current status of the plant or reactor safety as a whole. This increases the risk of human error.

<u>Solution for Mochovce units 3 and 4</u>

The solution for the control room should be on a completely different quality level because it will be designed for a new integrated control system. Its output equipment consists mainly of vision displays (EGP). The goal of the solution will be the formulation of requirements concerning the spatial solution and equipment delivery, that the result is a compact plan based on the functional analysis for dividing the function between man and machine and fitment of means for the contact with personnel. The output should be data to solve the conditions of the environment for the technical equipment and the personnel and to assess potential impacts of the environment and the personnel. Concerning the building's structure, the improvements concern the construction for a seismic fixing of the consoles and panels, providing an electric fire signalisation, lighting and cladding for the walls and ceilings.

### 3.2.2.23. Physical and functional separation of the control room and the emergency control room

In the event that an accident makes the control room unusable, the usability of an emergency control room has to be assured.
Fire or a short circuit in the control room potentially could result in a failure also in the emergency control room (or vice versa), and as a consequence, both control rooms are out of commission.

The two control rooms must be separated physically and functionally, taking into consideration the possibility of common mode failure. Up to now it is not clear whether this separation has been completed. No obvious mistake has been identified. However, the complex analysis which presents positive results is missing.

<u>Solution for Mochovce units 3 and 4</u>
(EGP) promises to separate the two control rooms physically and functionally, mainly under the aspect of electrical separation. However, it was not specified how this would be done. The available documentation indicates that alternatives are being considered, including one model where the emergency control room would be transferred into another object, no details on how this would be done

### 3.2.2.24. Primary circuit diagnostic system

The original project of VVER-440/213 reactors did not have any diagnostic system which would monitor potential threats to the primary circuit integrity and make a timely warning possible, when a defect at the pressure interface of the reactor coolant system occurs.
Without such a diagnostic system the operator has a very limited possibility to distinguish the presence of leaking parts of the reactor cooling system and the potential danger of a local core overheating or integrity loss at the pressure interface of the reactor coolant system.

The leak detection system gives a timely warning about the beginning anomalies at the pressure interface, like a damage of a main coolant pump seal or a small leak.

The existing system for monitoring impenetrability at the first circuit components with screwed joints was considered dysfunctional as a result of corrosion, clogging etc., because this makes it impossible to detect damage.

<u>Solution for Mochovce units 3 and 4</u>
(EGP) suggests in the framework of the measures called ´Monitoring of conditions for the environment of machinery equipment´ to add a fatigue monitoring system for the primary circuit; without any details given.

### 3.2.2.25. Reactor vessel lid leak monitoring

The bloc of protection tubes of the control assemblies drivers, the instrumentation etc., is connected to the nozzles through the reactor vessel lid with screwed joints (flanges). Each joint is sealed with two parallel nickel packing rings. The tube for collecting the penetrating water is of very small diameter, easily gets clogged and stops providing usable information as a consequence. No controls or tests of the leak detection system are performed. The humidity monitoring system in the upper part of the reactor bloc is not sensitive enough to detect flange leaks.
An unobserved leak of primary circuit coolant containing boric acid can lead to an extensive reactor vessel lid corrosion from the outside. The reactor lid is covered with a metallic structure which is filled with ceramic balls and therefore their upper layer is not such, that a routine check would detect corrosion damage.

Some vessel lids had to be repaired as a consequence of corrosion at NPP with VVER-1000.

<u>Solution for Mochovce units 3 and 4</u>
(EGP) suggests to re-evaluate some not further identified study with the goal to minimize the necessary equipment of the diagnostic system to the necessary scope (!). Further it promises a solution, which would fulfil the condition for the statute LBB. It seems that the concrete problem of monitoring reactor lid leaks is been solved in the framework improving the diagnostics of the primary circuit as a whole.

### 3.2.2.26. Post-accident monitoring

The post-accident monitoring is used to inform the operator about safety relevant parameters connected to the defence-in-depth concept. Such information is indispensable for a number of application procedures concerning safety and it reduces the probability of a faulty action.

Such a system is not available at an NPP with VVER-440/213.

<u>Solution for Mochovce units 3 and 4</u>
Also for this issue (EGP) considers it necessary to re-evaluate the monitoring parameters. The solution should principally be different from the concept applied at the first dual bloc. A qualified water level measuring in the reactor vessel should be added. The post-accident monitoring system will be solved as a part of the new I&C system.

### 3.2.2.27. Technical support centre

A much used method is to establish a spot, where current data about the plant is collected and the status of the plant to display it to the technical experts, who support the operators during accident states management. This room is separated from the control room.

Solution for Mochovce units 3 and 4
(EGP) does not solve this problem. We can assume, that the solution will be part of the complete exchange of the Instrumentation & Control System.

### 3.2.2.28. Plant power supply under accident conditions

The power supplied by the diesel generators is for the safety systems, which are indispensable to control a maximum design basis accident. The number of systems supplied by the diesel generators however, is limited compared to the international standard and the power supply does not include some systems, which serve to reduce the severe accident risk.

Example of safety relevant systems, which at NPPs with VVER-440/213 are not part of the diesel generator emergency supply:

- reversible packing water pumps for the main coolant pumps
- charging system for the batteries, which ensure an undisturbed power supply to the I&C systems and the technological computers,
- cooling system of the control assemblies,
- control panel for radioactivity monitoring,
- phone connection for communication between control room and plant,
- pumps for filling the fuel tanks for the diesel generators (tanks contain fuel for 8 hours of diesel generator operation)
- safety relevant systems in the machine hall.

The listed systems are necessary events under control, which were caused by power loss and the bloc had to be cooled down. The significance of the power supply to the reversible packing water pumps for the main coolant pumps is described in chapter 3.2.2.10. Moreover the functional reversible pumps in consequence speed up reaching the necessary concentration of boric acid in the shut-down reactor (coolant does not get lost).
The power supply of the mentioned systems cannot be guaranteed to come from the diesel generators, because their capacity is used up by other appliances.

This weak point has been identified during the operation of VVER-1000 reactor, especially the accident at the NPP Kosloduj (units 5 and 6) in September 1992 made this obvious.

Solution for Mochovce units 3 and 4
(EGP) suggests the preparation of an analysis and plans for the engineering part. The plan is to use diesel generators from the neighbouring bloc by connecting the

system distribution 6 kV II. category of secured power supply. The appliances should also be supplied by sources in the external grid.

(EGP) classifies this issue as a category I problem.

### 3.2.2.29. Discharge time of the emergency batteries

The batteries are the last energy source in a power plant and therefore high reliability and capacity are the main goals. Each plant with VVER-440/213 has three redundant batteries to ensure power supply for the most decisive appliances. Their discharge time according to design is usually 30 minutes. This situation is not in line with modern requirements. In the situation of total station black out the accumulators are the last energy source for the plant. Their capacity keeps the I&C and the lighting of the station in operation. It should make the monitoring of the basic plant parameters possible and the engine driven safety relevant valves should stay manoeuvrable. The reactor has to be kept under safe conditions. If the battery discharge time would be longer, this would give the operators more time to decide on further actions.

The advantage of the VVER-440 reactors lies in the fact that they can endure a longer time without power supply without causing core damage, because of the high amount of water in the primary and secondary circuit. To be able to make use of this advantage, the operators should have the plant under control for the time necessary until the plant´s power supply is re-newed. Many power plants exchanged the batteries to achieve a higher capacity, however, this measure was not implemented everywhere.

Another deficiency is the lack of an accumulator monitoring system. Moreover the batteries are not adequately insulated from the concrete floor and cannot cope with seismic loads. An earthquake can lead to battery loss and therefore to loss of continuous power supply.

<u>Solution for Mochovce units 3 and 4</u>
(EGP) suggests abolishment of the 24 V DC distribution. The new requirement for the delivery of the new I&C system should be including, that the supply of these systems and the connections in the electric part will be organised exclusively over a 220 V DC or 380 resp. 220 V AC. This should increase reliability, operational flexibility and internal redundancy of the systems with secured supply. The load for the secured supply system of category I should be reduced (e.g. the emergency lighting of the external non-seismic objects should be transferred to the sources of category III). The battery of the secured supply system should be designed to supply this load for a time of 120 minutes.
(EGP) classifies this issue as a category I problem.

### 3.2.2.30. Thermodynamic bubbler condenser behaviour

The information about the bubbler condenser behaviour under transients is insufficient. Additional information is needed as the basis for defining the weak points and for taking measures to eliminate them.

The thermo-hydraulic parameters about the activity of the Bubbler Condenser system were verified with separate tests on a reduced scale, which did not indicate any dangerous phenomena.

The experiences collected during the tests of the boiling water reactors and the confirmed event at a German plants showed, that the process of steam condensation is connected to effects, which lead to strong pressure oscillations and can damage long steam leading tubes under the water surface.

The geometric layout of the barbotage scuttles is different in the individual plants. The designers believe, that their small scale tests and their theoretical calculations confirm, that the above mentioned effects do not happen. The test results on bigger scale, however, are still lacking. Such tests would have to be conducted under conditions representing the conditions in a power plant and this could indicate an oscillation with a much higher pressure and interaction between medium and metal built-ins, which are dangerous for the integrity and function of the pressure relief condenser.

This load was not taken into consideration when the mechanics of the pressure relief condenser (walls) were designed. There is tendency of metal built-ins to collapse under LOCA accident conditions, which can lead to the collapse of the whole system.
It is necessary to conduct a full-scale thermo-hydraulic experiment to prove, that unpredictable pressure oscillation and interaction between medium and metal built-ins, which could put the pressure relief system integrity at risk during accidents, do not occur. The power plant operators however, point out that the system design is sufficiently conservative and do not admit that effects can occur, which will seriously endanger the system functioning.

Solution for Mochovce units 3 and 4
(EGP) does not solve this issue.

### 3.2.2.31. Non-leak tightness of the hermetic zone

At the majority of plants with VVER-440/213 the leak values from the hermetic zone reached around 100% volume per day with a maximum overpressure 0,25 MPa. Even though this value is under the limit, a reduction is necessary to improve the environmental protection under accident conditions. This value could be reduced by using more sensitive methods and equipment to detect leaks during tests and by an improved quality of work.

Solution for Mochovce units 3 and 4
(EGP) does not deal with this issue.

### 3.2.2.32. Maximum pressure difference at the walls between the individual boxes of the hermetic zone

It is necessary to confirm the stability of the internal walls and ceilings of the hermetic boxes, to exclude the risk of destruction of the safety relevant equipment (measuring devices, cabling, impulse tubes) as a consequence of wall or ceiling collapse. No analyses concerning the pressure differences were conducted.

Solution for Mochovce units 3 and 4
(EGP) does not deal with this issue.

### 3.2.2.33. Systematic fire risk analysis

Fire risk analyses are necessary to verify the localisation and separation of safety relevant equipment, the needed fire resistance at the fire compartment borders, the requirements for the fire extinguishing equipment and further issues necessary to fulfil the fire protection requirements. For power plants under construction it is necessary to prepare those analyses before start-up. Power plants in operation should prepare such analyses regularly.

Solution for Mochovce units 3 and 4
The risk analysis for the second dual bloc will be conducted only after the situation of the 1. and 2. bloc was analysed (EGP). The review for bloc 3 and 4 should include the partial difference in the fire protection systems, mainly concerning the cabling solutions.

### 3.2.2.34. Fire detection and fire extinguishing

Safety analyses of power plants with VVER reactors identified some weaknesses in the area of fire detection and fire extinguishing, which count as deviations from the international standards.

One of those issues is the functionality of the fire detection and fire signalisation system under abnormal conditions. The design of the detection system and fire signalisation was designed according to conventional industry standards without the ability to withstand earthquakes or other abnormal conditions characterized by mechanical, thermal, chemical or other effects, which can occur as the consequence of design basis accidents. In case of such abnormal conditions the system might not be able to detect the fire or set an alarm.
The second issue concerns the activation of the extinguishing water supply system. The design of power plants with VVER-440/213 has three independent fire extinguishing water lines. If this line fails, it is necessary to activate the other lines manually. The requirements (including the Soviet requirements) foresee a simultaneous activation of all extinguishing water lines.

Another issue is the fact, that the control room and the emergency control room and other rooms with I&C equipment with electronic and electric devices of a surface of more than 20 $m^2$ do not have an automatic gaseous fire extinguishing system. This is not in line with the requirements (including the Soviet ones).
In the original design the fire protection detection and the fire suppression system were separate in the rooms which they shared with the main coolant pumps and reversible pumps.

Solution for Mochovce units 3 and 4

The existing electronic fire signalisation is to be exchanged against the ESSER equipment including plans to install fire extinguishers in the individual buildings of the power plant and connecting them to the new central fire offices in the inoperative part of the control rooms and in the fire station (EGP).

The extent of the fire which could possibly occur, should be limited with some measures in the reactor protection system rooms and (EGP) suggests to consider (!) local fire extinguisher system for the individual boxes. Fires which affect the boxes with electronics should be extinguished with stable fire extinguishers. Also the installation of fire extinguisher equipment for the transformer stations should be improved ant the fire-extinguishing agents used for the turbo generator oil tanks exchanged. The fire hydrants should be connected to seismically resistant distributors.

At the second dual bloc a main coolant pump of the new generation without an oil system should be installed, which would take away one of the fire risks.

The activation of the extinguishing water is not being solved.

### 3.2.2.35. Mitigation of the consequences of a fire

The safety analysis of power plants with VVER identified some weak points in the area of fire consequences mitigation, which show some deviations from the international standards.
One issue concerns the source of extinguishing water for use inside the hermetic zone. In some power plants the extinguishing water for those systems is taken from the non – essential service water system rather than the essential service water system.

This represents a deviation from the requirements (including the Soviet ones). Another problem is that the rooms with potential fire risk and the evacuation corridors do not have a system to remove fume in case of fire. This could put the personnel at risk and lead to problems when evacuating the personnel. Again, this represents a deviation from the requirements (including the Soviet ones).

The steel ceiling of the machine hall has limited capacity to resist the impact of a substantial fire in the machine hall. Under the impact of heat development, this construction loooses its mechanic strength and the ceiling can collapse and put safety- relevant equipment at risk. Accidents with the breaking out of a fire in the machine hall show the importance of having fire resistance steel constructions built into the ceiling.
The corridor on 14,7 m, where the steam generator safety valves are located, together with the separation valves on the steam pipe and the regulation valves of the steam generator feeding system, form one shared room with the machine hall. In case of a fire in the machine hall, there is no protection of this corridor. Temperature in this corridor rises over the course of 10 minutes up to 120 – 140° C as a result of the high temperature flow. Such high temperatures could lead to the damaging of the equipment located here. This can put even the reactor cooling system at risk.

Solutions for Mochovce unit 3 and 4
Sufficient exchange of air should be guaranteed at the protected leak routes (EGP). In the air conditioning distribution, fire flap valves should be installed to prevent fumes from spreading into the rooms that are not affected by the fire. A system to remove fumes and heat from the machine hall in case of a fire there should set up.

The turbo generator oil tank should be equipped with a covering, to limit the spreading of fumes during a fire. In the wall on the 14,7 m floor measures should be implemented to eliminate the impact of a fire on the equipment of the safety system.

### 3.2.2.36. Human induced external event

Nuclear power plants with VVER-440/213 are vulnerable towards external impact. This is due to the fact that the monolithic concrete part of the reactor hall is designed to withstand extreme internal events and that there is no external containment construction to resist external impacts. The power plant site is usually chosen so that the probability of a plane crash or an external explosion is low. However, no complete analysis has been conducted for all individual power plants. To reduce the probability of similar events, administrative measures can be taken.

Solution for Mochovce units 3 and 4
(EGP) does not directly solve this problem. It can be assumed that a solution will come out of the completed probability safety analysis, which (EGP) actually has demanded should be performed.

### 3.2.2.37. Scope and method of accident analysis

The accident analyses prepared earlier do not reach international standard. Those original analyses were reviewed in the years 1991-1992. The scope of the analysed accidents was extended as well as the time frame for which the calculations were made, and the quality of work improved.

Even though nuclear power plants with VVER-440/213 are very similar, it is not possible to transfer the accident analyses from one to another. It is therefore necessary to verify whether the accident analyses are complete for each individual plant. Also, a systematic overview of analytical methods is necessary.

Solution for Mochovce unit 3 and 4
(EGP) does not directly solve this problem. It can be assumed that a solution will come out of the completed Probability Safety Analysis, which (EGP) actually has demanded should be performed.

### 3.2.2.38. Codes and validation models of power plants

The results of accident analyses are reliable only if analysis was made with codes and models validated for the respective accident and the respective plant. The validation of codes and models cannot be assessed separately. It has to be undertaken for the respective combination of codes and models.

The applicability of codes used for accident analysis of Western reactor types to assess VVER reactors will probably not be a problem. The basic physical effects are not differing. The range for which certain correlations are used, however, can be different for both types and the validity of the correlations in the VVER range has to be checked.
However, there are important differences in the construction of VVER and Western type pressurized water reactors. Those differences require specific modelling techniques. This mainly concerns:

- horizontal steam generators
- the fuel part of the control assemblies
- hexagonal geometry of the fuel
- fuel element cladding

Solution for Mochovce units 3 and 4
(EGP) does not directly solve this problem. It can be assumed, that a solution will come out of the completed probability safety analysis, which (EGP) on the other hand demands.

### 3.2.2.39. Transient states with under cooling in relation to a pressurized thermal shock

If a transient process with an under-cooling starts at a time, when the reactor vessel already has been exposed to a high neutron flux and the pressure in the primary circuit is fairly high, the probability increases that the delivered cold water causes a thermal shock in the lower weld of the reactor pressure vessel. This causes a threat to the vessel integrity as a consequence of its embrittlement. The risk increases with the aging of the power plant and the reactor vessel embrittlement.

Solution for units 3 and 4 Mochovce
(EGP) does not directly solve this problem. It can be assumed that a solution will come out of the completed Probability Safety Analysis, which (EGP) actually has demanded should be undertaken.

### 3.2.2.40. Steam generator collector break analysis

A steam generator collector break is an accident with a high safety-relevance for two reasons in the case where the shutdown of the steam dump to atmosphere (PSA) does not close:

- hermetic zone by-pass and leak into the environment via the steam dump to the atmosphere (PSA):
- risk to the long term core cooling in case of loss of coolant via PSA.

Solution for Mochovce units 3 and 4
(EGP) does not directly solve this problem. It can be assumed that a solution will come out of the completed probability safety analysis, which (EGP) actually has demanded be undertaken.

### 3.2.2.41. Low power and reactor shut-down accident

When the reactor is shut down for maintenance or refuelling, some important safety systems are turned off or isolated. Moreover, the operator is expected for several reasons to perform a high amount of actions. From a safety point of view, this means that there are less barriers and less protection against events that can lead to an accident.
The analyses on the situation during shut-down and during fuel exchange at power plants with VVER-440/213 are not available or exist only partially.

Results of studies done worldwide show that the accident risk in the phase of reactor shut-down and fuel exchange is high. A significant contribution to this risk is the potential dilution of boric acid, loss of residual heat removal system under conditions of reduced medium, primary circuit loss, loss of power supply, fire and human error.

Solution for units 3 and 4 Mochovce
(EGP) suggests to conduct a comprehensive safety study about accidents during low output and reactor shut-down. The risks are to be assessed by applying a deterministic approach. In the next step, a Probabilistic Safety Analysis for the operation at low output and during reactor shut-down should be performed.
(EGP) does not classify this problem as category II.

## Conclusions

The first step to assess the possible nuclear safety level for the planned completion of units 3 and 4 at the Mochovce nuclear power plant is to define the standard that can be applied to the suggested measures. For power plants with VVER-440 reactors, it is not correct to compare the relatively specific capacity class and age of the design to current reactor types. Even the Slovak Nuclear Regulator (UJD) and the project planner (EGP) admit that they cannot reach the current level.

For the Mochovce 3 & 4 project, the IAEA classification of safety problems of VVER-440/213 reactors is used as standard. IAEA documents represent an authority in the nuclear energy field and the cited document is comparatively critical towards VVER reactors. It is of course possible to discuss the question of whether the categorization of individual problems sufficiently describes their significance for nuclear safety. After all, even (EGP) does this and in some cases it assigns some problems into categories according to their own judgement. This study avoids this discussion by dealing with the most important categories equally.

The only available serious document describing the planned measures to increase nuclear safety is the concept (EGP). Its disadvantage for the purposes of this study is that it is not a very elaborate plan on how to solve individual problems. A number of the suggestions only relate to the analysis of the respective problem and suggests that the concrete measures will only come out of the analysis. Therefore, it cannot be excluded that the concrete measures in

the end will not fulfil the requirement of the categorisation (IAEA). This supposition is confirmed with the document's tendency to re-evaluate some solutions to save on costs (shared sensors, scope of measured parameters, seismic measures etc.).
This study therefore represents a first approximation. To achieve an approximation of a higher level it would be necessary to have available more concrete solutions to the problems. More suggested solutions could in turn point to more deficiencies, which currently are not recorded even by (IAEA), nor (GRS).

However, even this first approximation concludes that the planned completion of units 3 and 4 at Mochovce does not even reach the standards required for VVER-440/213 classification (IAEA).

Out of **eight** IAEA category III issues, the plan (EGP) (with an uncertainty stemming from the non-elaborated plan) solves the following:

- **two issues completely** – non-destructive testing, seismic measures

- **five issues partially** – equipment qualification, sump net packing of the emergency core cooling water system, feed water supply reliability, fire prevention, high energy pipe break risk

- **one issue not addressed –** Bubbler Condenser behaviour at maximum pressure difference after LOCA accident.

Out of **forty-one** IAEA category II problems, the plan (EGP) (with the uncertainty stemming from the non-elaborated plan) solves the following:

- **28 issues completely** - Component classification, reliability analyses for the systems in Safety Class 1 to 2, pressure vessel integrity, primary circuit protection against cold pressurizing, mitigation of the consequences of the steam generator primary collector break, qualification of pressurizer relief and safety valves for water flow, integrity of suction pipe systems of the ECCS, heat exchanger integrity in the ECCS, qualification of steam generator relief and safety valves for water flow, control room ventilation, hydrogen removal system, Instrumentation and Control system (I&C), overview over signals initiating a reactor scram, ergonomics of the control rooms, physical and functional separation of the control room and the emergency control room, primary circuit diagnostic system, reactor vessel lid leak monitoring, post-accident monitoring, technical support centre, plant power supply under accident conditions, emergency batteries discharge time, systematic fire risk analysis, mitigation of the consequences of a fire, scope and method of accident analysis, codes and validation models of power plants, steam generator collector break analysis, low power and reactor shut-down accident

- **6 issues partially** - Preventing unintended dilution of the boric acid (analysis), main coolant pipe whip, primary steam generator collector integrity (analysis), primary steam generator collector integrity (analyses), steam generator tubes integrity (analyses), fire detection and fire extinguishing

- **7 issues not addressed** - Functioning of steam generator relief and safety valves under low pressure conditions, primary circuit ventilation under accident conditions, thermodynamic bubbler condenser behaviour, non-leak tightness of the hermetic zone, maximum pressure difference at the walls between the individual boxes of the hermetic zone, transient states with undercooling in relation to a pressurized thermal shock

The most striking is that the problems regarding Bubbler Condenser System reliability and equipment protection against external events (lack of a containment) are ignored. An explanation might be the practical impossibility to implement principal improvements with acceptable costs.

As was described in chapter 1, not even the full implementation of IAEA requirements and recommendations would bring the Mochovce nuclear power plant up to a safety level that would satisfy the requirements resulting from the fact that there are more than 400 reactors worldwide that are under operation for 12 000 reactor-years. The potential completion of units 3 and 4 would therefore contribute to the risks, even significantly, which operators of nuclear power plants wilfully and for profit’s sake burden society with.

## Abbreviations

| | |
|---|---|
| EGP | Energoprojekt |
| ECCS | Emergency Core Cooling System |
| NPP | Nuclear Power Plant |
| LBB | Leak Before Break |
| LOCA | Loss of Coolant Accident |
| SE | Slovenske elektrarne |

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(HEŘMANSKÝ) Heřmanský, B.: Těžké havárie tlakovodních reaktorů s tavením paliva, ÚJI, Praha, prosinec 1990 *(Severe accidents with fuel melt at pressurized water reactors, ÚJI, Prague, December 1990)*

(SE) www.seas.sk/elektrarne/jadrove-zariadenia/

(SE&EDF) Anonymus: Program zvýšenia jadrovej bezpečnosti v Slovenskej republike, zvýšenie bezpečnosti a dokončenie blokov 1 a 2 Jadrovej elektrárne Mochovce, Slovenské elektrárne, Electricité de France, Bratislava, Paris, december 1994 *(Programme to increase nuclear safety in the Slovak Republic, increase of nuclear safety and comopletion of units 1 and 2 NPP Mochovce, SE, EdF, Bratislava, Paris, December 1994)*

(GRS): Anonymus: Sicherheitsbeurteilung des Kernkraftwerks Greifswald, Block 5 (WWER-440/W-213), GRS-83, Gesellschaft für Reaktorsicherheit mbH, August 1991 *(Safety assessment of the NPP Greifswald, unit 5(VVER -440/W-213), GRS-83, German reactor safety research institute GRS, August 1991)*

(IAEA) Anonymus: Safety Issues and their Ranking for WWER-440 Model 213 Nuclear Power Plants, IAEA-EBP-WWER-03, International Atomic Energy Agency, April 1996

(EGP) Anonymus: Koncepce dostavby 3. a 4. bloku JE Mochovce, Energoprojekt Praha, a.s., Praha, červenec 2002 (*Concept for the completion of units 3 and 4 NPP Mochovce, Energoprojekt Praha, a.s., Prague, July 2002)*

(ÚJD) Šimončič, A.: Stavebné povolenie – zmena podmienky, Úrad jadrového dozoru, Trnava, 8.7.2004 (*Construction permit – change of conditions, UJD – Nuclear Safety Regulator of SR, Trnava, 8.7.2004)*

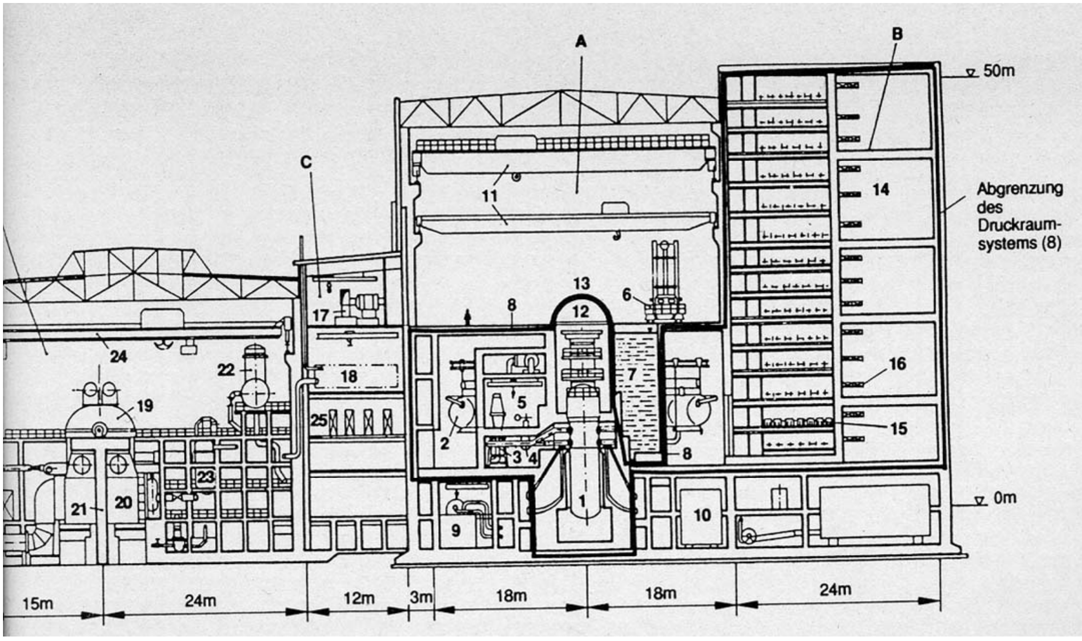
(MPO) Anonymus: Informace o průběhu modernizačního programu v Jaderné elektrárně Dukovany, Ministerstvo průmyslu a obchodu, Praha, červen 2001 (*Information about the mondernization programme at NPP Dukovany, Ministry of Industry and Trade, Prague, July 2001*)

**Annex 1**

## Function of the Bubbler Condenser System

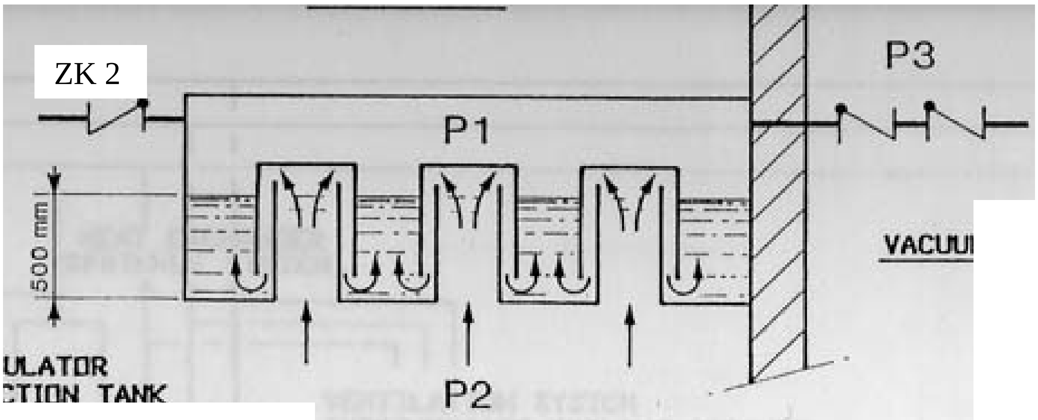
The reactor circulation loops, which bring the cooling water into the core and circulate it back, are placed in the so called hermetic boxes, connected with a channel to the Bubbler Condenser Tower (position B in figure P1). The bold broken line is marking the system pressure boundary in figure P1. These hermetic boxes are under low pressure and regularly being checked for tightness.
The maximum design basis accident is the complete break of the main circulation pipe with coolant flowing out on both ends of the ruptured pipe. As a consequence of the sudden pressure loss in the primary circuit, radioactive steam would immediately start flowing out of the damaged loop and cause a fast pressure increase in the hermetic rooms. The destructive impact of this event is supposed to be prevented by the connection of the hermetic rooms with the Bubbler Condenser tower via the channel. The steam-air mix flowing from the pressurized hermetic boxes condenses while going through the trays, filled with boric acid solution (position 15 in figure P1). Non – condensed gases are gathered in gasholders (position 14 in figure P1).

**Figure P1: Section through a plant with a VVER-440/213 (GRS)**




The barbotage system consits of 12 barbotage trays filled up to the level of 500 mm with boric acid and water solution (12 g/l) and hydrazine (0,1 g/l). In total the systems contains 114 $m^3$ solution with a temperature of 35°C – 40°C. One room with three trays each is connected with a clack valve to one gasholder chamber. The gathered gas is later released via the filtered venting system into the atmosphere.
The system also consists of a pump to remove the solution from the trays for cleaning and the solution cooler. According to the limits and conditions the system can function with 11 trays filled up to 500 mm.

**Figure P2: Scheme of the barbotage system functioning** (SE&EDF)




When the pressure difference $p_2$-$p_1$ is under 5 kPa, the level in the water seal drops and rises in the tray. If this difference exceeds 5 kPa, the water seal is blown through, the steam-water mix from the hermetic boxes bubbles through the solution, cools down and condenses over the water level, which causes the pressure to rise in this room (p1). When the pressure difference $p_1$-$p_3$ reaches values over 0,5 kPa, the clack valve opens up and the non-condensed gases are releases into the gasholder.
In the next stage of the post-accident sequence, the sprayer system starts up, which makes the pressure inside them go down. As soon as the pressure difference $p_2$-$p_1$ reaches less than 5 kPa, the bubbling through of the steam-water mix is interrupted, the pressure difference $p_1$-$p_3$ drops and the clack valve closes. Non-condensed gases get caught in the gas holder.
If the pressure $p_2$ continues to drop and the difference of the pressures $p_2$-$p_1$ sinks under zero and at the same time $p_2$ exceeds 165 kPa, the solution pours out in the trays (spraying the Bubbler Condenser Tower during the time the pressures stabilize in the hermetic zone and the barbotage scuttles). While this pressure lasts (the pressure in the hermetic boxes), the opening of the clack valve ZK 2 is blocked (see figure P2). When the pressure is under 165 kPa (and pressure $p_1$ is higher than $p_2$), the room with pressure $p_1$ looses pressure via the clack valve ZK 2. As a consequence the solution does not pour out in the tray.
A design basis accident does not assume pressure to exceed 245 kPa.

(GRS) as well as (IAEA) make reservations about the functionality and reliability. They are summarized in chapter 3.2.1.5. and 3.2.2.30. They share the main reservation: the system was not tested full-scale and its reliability is only deducted from model calculations and partial tests conducted on a reduced scale.

Furthermore it is obvious, that the function of a Bubbler Condenser is not and cannot be to protect the nuclear installation against external events. However, some documents still claim, that the Bubbler Condenser replaces the containment. A Bubbler Condenser is a passive safety system, which serves to reduce pressure in the hermetic boxes after a circulation pipe accident and to catch non-condensable radioactive gases after a primary circuit accident. It is not comparable to a containment, which also serves to protect the reactor from external impacts.

**Annex 2**

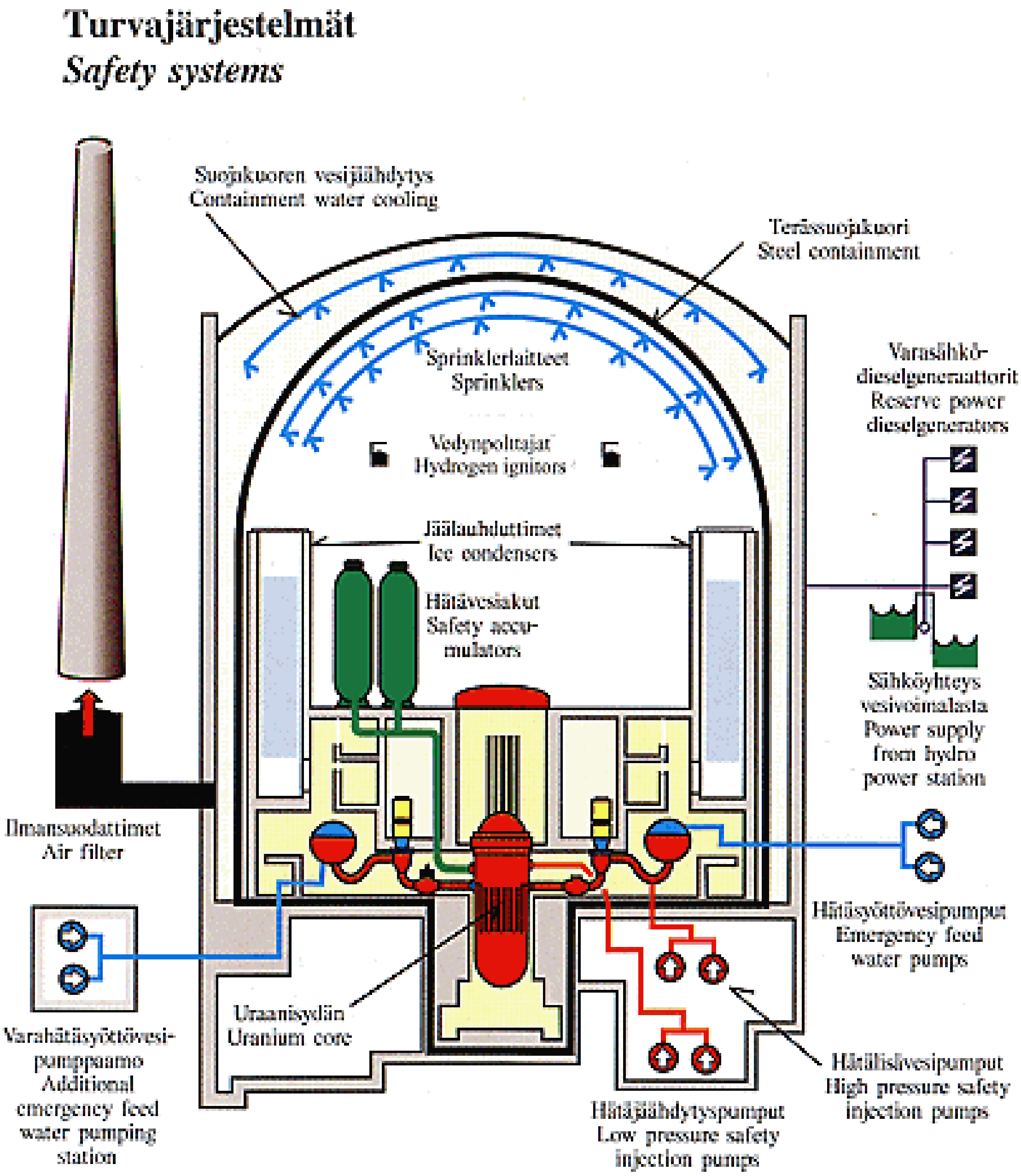
**Two examples of unusual solutions for nuclear power plants with VVER-440/213**

**Nuclear power plant Loviisa**

The only nuclear power plant with a VVER-400/213 reactor which has a containment is the Finnish nuclear power plant Loviisa. Its two reactors were equipped with an ice condenser by Westinghouse and an Instrumentation & Control System by Siemens; reactor start-up was in 1977 and 1980. The reactor, the turbines, generators and other main components were delivered from the Soviet Union. The share of domestic delivery was around 50%.

A special solution to the containment is not only the ice condenser, but also its double-shells, where the inner steel shell is being cooled from the interspace (see figure P4). Such a solution was not even applied at the third generation VVER (VVER-1000)! The inner gastight steel shell is protected against external threats by a ferro-concrete shell.

**Figure Nr.4: NPP Loviisa Safety System**
(http://www.fortum.com/document.asp?path=14022;14024;14026;14043;24939;29154;29452)




## Nuclear power plant Greifswald

Four units with VVER-440/213 reactors were also built in the former German Democratic Republic at the site Greifswald (Lubmin).
After the German re-unification investors assessed the safety increase necessary according to German regulations in combination with the than overcapacity in the German market as too costly: in 1990 the construction of three units with VVER-440/213 was stopped and the units in operation (4 units with VVER-440/230 and 1 unit with VVER-440/213) stopped. Certainly the protest of environmental and human rights organisation at the site and the fire at unit 5 on November 24$^{th}$ 1989 increased the pressure to take this decision. The accident was caused by a test bypassing of the electric circuit, where a shortcut caused a fire in the cabelling. An accident similar to the one on 7$^{th}$ December 1975 at the first unit, where for the same reason a fire broke out, led to the destruction of the power supply and the control of five main feedwater pumps.
Since 1995 the plant is being decommissioned.