umweltbundesamt

Report to the Austrian Government on

Paks NPP Lifetime Extension

Environmental Impact Assessment

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Introduction

In September 2005, the "Statement on the Preliminary Impact Assessment Study" was prepared by the Umweltbundesamt on behalf of the Austrian Government. That Statement contained a number of requests for information.

The below expert statement investigates to which extent those requests have been met by the information provided in the "Environmental Impact Assessment Study" and the "Answers to the study of Umweltbundesamt".

Furthermore, the authors discuss the effect of lifetime extension of Paks NPP on the safety status of the plant and on the potential accident hazard for Austria. Since the safety assessment of Paks NPP is ongoing and several modifications are in the planning stage, the authors identify the fields of interest for which follow up of the development and/or further discussion is required in order to assess the risk of severe accidents in Paks NPP.

This statement refers to all documents Austria received from Hungary within the framework of the EIA process for the lifetime extension of Paks NPP:

- the **Preliminary Environmental Study (PES),** 2004
- the **Ruling** concerning the environmental licensing of the Paks NPP life extension**,** herein after referred to as the **Ruling,** 2005[1]
- the **Environmental Impact Study (EIS),** 2006
- the **Answers to the study of Umweltbundesamt** to the statements of the provincial authorities of Lower Austria, Burgenland and Wien, Greenpeace, Global 2000 et al., herein after referred to as the **Answers**, 2006

and to the

- Report to the Austrian Government, EIA procedure on the lifetime extension of Paks NPP - **Statement on the Preliminary Impact Assessment Study,** Umweltbundesamt, Vienna September 2005 - herein after referred to as the **Austrian Statement**

During the discussion at the public **Hearing** of June 6, 2006 in Mattersburg the Hungarian side provided some new information regarding the issues brought up by the Austrian experts. After the Hearing this Statement was completed by a discussion of the new knowledge from the Hearing.

This statement is divided in two parts. Part I deals with 8 technical issues chosen because of their importance to the lifetime extension, and Part II deals with the accident risk and potential impact on Austria.

1 It is noteworthy to mention, that it is unknown whether the competent environmental body had released amendments to the Ruling considering the requests of the Austrian Statement.

Summary and Conclusions

In the **Austrian Statement** on the Preliminary Environmental Study , questions were raised which are of importance for the risk of extended plant operation and therefore connected with the issue of severe accidents.

According to the **Environmental Impact Study** for Paks NPP, the total core damage frequency (CDF) is 3.0 10^{-4} per unit and year. The dominating contribution, with 86% of the total, is the seismic risk.

The value of 3.0 10^{-4}/year is considerably higher than the target for severe core damage frequency for existing nuclear power plants set by the IAEA's International Nuclear Safety Advisory Group (1.0 10^{-4}/yr) [INSAG 1999].

The core damage frequency of 2.58 10^{-4} per unit and year for seismic events alone constitutes a very high value. An earthquake will hit all 4 units at once and therefore can lead to severe accidents in all four reactors simultaneously. Moreover, all other facilities at the site, including the spent fuel store, would be endangered by an earthquake as well. After an earthquake, the situation at the plant could be extremely complex and confusing.

At the public Hearing in Mattersburg the Hungarian representatives presented their latest results of safety analysis:

- due to the first reconstruction measures in Paks NPP a reduction of seismic risk to a core damage frequency of 6.6 10^{-5}/year was achieved. Although this is a substantial improvement, the units of Paks NPP are only just meeting the IAEO target value of 1.0 10^{-4}/year.

Regarding beyond design base accidents the Hungarian side explained that 80% of the analyzed BDBA – sequences lead to a cesium-release of less than 1% of the inventory and only 6% to a release of more than 20%.

In this statement, various questions are raised concerning potential problems with safety relevance, which further emphasize that beyond design base accidents are possible. Thus, it is clear that the discussion of potential transboundary effects cannot be restricted to design base accidents.

The release of radioactive substances can affect regions in a distance of several 100 km from the source. The **Austrian Statement** of September 2005 presented the assessment of potential impacts of a beyond design base accident for different weather situations in Central Europe. In this assessment, the release of 30% of the Cs-137 inventory of one reactor core was assumed. In case of an earthquake or terror attack, the release could even be higher.

Below is provided an overview of reactions by Paks NPP in the **Answers** and treatment of the technical issues raised by the **Austrian Statement** in the **Environmental Impact Study**, followed by an assessment and recommendations for the follow up:

Ageing Management Program

In the **Austrian Statement,** information on the ageing management program was requested. According to the explanations in **Answers** and the **Environmental Impact Study,** this program is not yet completed, particularly not in the context of lifetime extension.

Further information concerning Ageing Management, focusing on the underlying regulation, was provided at the **Hearing**.

Assessment

Considerable changes and developments are to be expected in the Ageing Management Program of Paks NPP during the next years. Further information on this process is of high importance for the assessment of severe accident risk. As an important first step, provision of the information reported orally at the Hearing by the Hungarian side in written form would be helpful.

In particular, further observation should permit to ascertain that the new approach to in-service-inspections to be introduced at Paks, which is to include reductions in inspection efforts without a decrease of the safety level, indeed does not lead to any safety level decreases.

Reactor Pressure Vessel Ageing

Information concerning various material data, in-service-inspection, thermo-hydraulic and fracture mechanics analyses and counter-measures were requested by the **Austrian Statement.**

The safety assessment of Paks reactor pressure vessel is yet in an early stage. Data given in the **Environmental Impact Study** are from Loviisa. In the **Answers** provided by Paks NPP, the ageing management of the pressure vessel is briefly described. It is emphasized that a complete new safety and component ageing analysis is required for licensing the lifetime extension.

Aspects not discussed in **Environmental Impact Study** and **Answers** are the dose rate effect and the effect of changes of emergency core cooling system. Of particular interest is the question of the safety margins which are to be applied in the pressurized thermal shock analyses. This point is only mentioned in passing, for Loviisa, were the margin is only 4℃ instead of 10℃, as recommended by the “Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants” [IAEA 1997a],

Assessment

The safety assessment of the Paks reactor pressure vessels, in connection with lifetime extension, is in an early stage. Even at the present stage, some information should already be available which would be of great interest from the Austrian point of view. This concerns for example the database for un-irradiated materials, a comprehensive description of the surveillance program, information on the scope of the thermal shock analyses, the thermo-hydraulic calculations already performed, and the methodology to be applied regarding fracture mechanics.

Of particular interest is the question of the safety margin to be applied in the thermal shock analyses, as well as assumption on cracks, on pressure vessel cladding integrity, and on the manner of warm pre-stressing effect application. Furthermore, the possible effects of a dose rate effect and the way this effect is investigated in the surveillance program would be of great interest for accident risk assessment. It would also provide proof for the claimed reduction of neutron fluence in the vessel wall in spite of power uprating.

Possible consequences of changes in the emergency core cooling system for the sequence of accidents would also be of interest.

As the new thermal shock analyses will be performed in the coming years, further information on methodology and results would be of great interest to the Austrian side.

Steam Generator Ageing

The **Austrian Statement** pointed out that corrosion of steam generators is an important issue for VVER plants and has created considerable problems at Paks NPP in the past. Therefore, information concerning the causes of the accident in 2003, particularly regarding aspects relevant to corrosion problems in the primary cooling circuit is of importance.

In the **EIS and Answers**, summary information on the Paks steam generators is provided, in particular regarding plugging and corrosion issues. The accident of April 2003 is discussed at some length in the EIS, without, however, addressing the underlying problem of steam generator corrosion.

At the **Hearing**, significant information was provided regarding scope and development of steam generator inspections as well as regarding steam generator tube plugging criteria.

Assessment

The safety assessment of the Paks steam generators, in connection with lifetime extension, is still in an early stage. The inspection system is still under development.

Provision of the information reported orally at the Hearing by the Hungarian side in written form could be helpful for the clarification of some remaining open questions (regarding stress corrosion cracking of tubes, erosion-corrosion of the feedwater system and follow-up work on the accident in 2003 regarding SG corrosion issues).

Information on the further development of in-service-inspections, particularly of possible modifications of the practice in the coming years, would also be of interest.

Confinement Ageing and Capability

The Austrian Statement suggested that potential ageing problems need to be dealt with, in particular the long-term behavior of the steel liner of the confinement rooms and the problem of high leak rates.

In **EIS** and **Answers,** information is provided on ageing management of the confinement system. Not discussed are: backfitting measures in connection with the PHARE experiments and investigations of the behavior of the confinement system in beyond design base accidents, its safety reserves and capabilities for accident mitigation.

At the **Hearing**, further information was provided for some aspects, mostly concerning leak rates and the situation in the confinement during DBAs.

Assessment

Because of the importance of the confinement system for plant safety, more detailed comments on ageing of the barbotage condenser system as well as on backfitting measures which were performed in the last years, or are planned for the immediate future (particularly in connection to the PHARE experiments and investigations, and the reduction of leak rates achieved in the last years), would be of interest to the Austrian side.

Also, a discussion of the behavior of the confinement system in case of a severe accident (as well as for variations of certain DBAs), including a discussion of safety reserves and of capabilities for accident mitigation, should be provided, as well as the consequences of the leak rate for the timing and extent of releases during beyond design base accidents.

As far as can be concluded from the available literature, there have been no comprehensive and systematic tests and investigations (such as have been performed for DBAs) into the capabilities of the confinement in case of beyond design basis events. In particular, it appears that no investigations have been performed regarding the behavior of the bubble condenser, which constitutes the critical part of the confinement system.

If such investigations have been performed or are planned nevertheless, they would be of great interest. A detailed discussion of planned and possible backfitting measures to improve the mitigating capabilities of the confinement system in case of BDBA should also be provided.

Seismic Hazard

The **Austrian Statement** emphasized the importance of seismic events and the requirement of reassessment of site seismicity and seismic design of the plant because of the rapid development of this scientific field and the implementation of new international guidelines and regulation.

The **Environmental Impact Study** confirms the necessity of dealing with the seismic issues: As the only external factor which can potentially lead to severe core damage, earthquakes were investigated in the probabilistic safety analyses. According to the results presented in the EIS, this factor is the dominating contributor to the core damage frequency: The overall core damage frequency is given as 3.0×10^{-4} per year and unit; 86% of this value are due to earthquakes. The **Answers** provided by Paks NPP describe, in some detail, the activities on the assessment of seismic hazards until today.

According to the results of very recent analyses briefly reported at the **Hearing**, the seismic contribution to CDF has been reduced to one quarter compared to the value given in the EIS, remaining, however, the dominant contribution among all events considered.

Furthermore, it is emphasized in the **Answers** that a new assessment of seismic hazards will be performed independently of the planned lifetime extension, in the framework of the next Periodic Safety Review (PSR), which is to be carried out for Paks NPP from 2006 to 2008. Backfitting has already been implemented and is being continued in order to increase resistance of buildings, systems and components to seismic events.

<u>Assessment</u>

New investigations of seismic issues, including a new assessment of seismic hazards, will be performed in the coming years. According to the still considerable contribution of seismic events to the overall risk at Paks, it will be of interest from the Austrian point of view to closely follow those investigations and assessments.

Seismic backfitting activities, too, are of considerable importance regarding the accident risk and should therefore be followed in the upcoming years.

The value given at the Hearing for the core damage frequency due to seismic events should be explained and discussed in more detail. Information on the current state of seismic backfitting would be of great interest from the Austrian point of view, as well as a discussion and an estimate of the reduction of core damage frequency to be achieved by further backfitting.

Terror Attack

In the Austrian Statement, it was emphasized that a terror attack against Paks NPP could have consequences for the Austrian population. It was pointed out that, as far as it is known, the reactor buildings at Paks NPP are not designed against the crash of even a small airplane, implying a high vulnerability to terror attacks.

The **Environmental Impact Study** does not contain any discussion of the issue of terror attacks. Furthermore, external impacts like an airplane crash or explosions are regarded to be very unlikely and hence are not considered either. The **Answers** provided by Paks NPP emphasize that the NPP meets the legal requirements concerning physical protection. No details are provided.

<u>Assessment</u>

Paks life extension is the first licensing procedure for a VVER 440/213 reactor since the attacks of September 11, 2001. The issue of terror attacks is discussed worldwide and should be discussed also in the procedure of Paks lifetime extension. Considering the grave consequences such an attack could have, and the current increase of the general terrorist threat in Europe, acknowledged by the Answers detailed information should be provided.

Vulnerabilities, attack scenarios and potential consequences can and should be discussed in an appropriate general manner, and in an appropriate setting. Regarding public debates, the criterion applied should be that it would be pointless to attempt to keep secret information which a competent group of attackers can easily acquire.

Power Uprating

In the **Austrian Statement** the question of the influence of power uprating on plant safety and thus on lifetime extension was raised. Concerns were expressed regarding reduction of safety margins, increased fuel corrosion, acceleration of ageing processes and new type of fuel, probably connected with higher burn-up and thus greater radioactive inventory. Furthermore, power uprating leads to acceleration of accident sequences in case of beyond design base accidents.

In the **Environmental Impact Study** a new subsection dealing with power uprating has been added. It was considered necessary to coordinate the two projects of power uprating and lifetime extension. In this context, Paks NPP commissioned a feasibility study from VEIKI AG concerning the effects of a power uprate on the ageing processes of the main components of the units. The result of this study was that power uprating would accelerate ageing processes; several modifications are mentioned in the documents which shall compensate or reduce the negative effects of power uprating.

At the **Hearing,** no further information concerning power uprate was provided. It was reported that the Safety Report describes the results of all analyzed DBA and BDBA sequences, but it did not become clear which accident sequences were performed on the increased power level.

Assessment

More information concerning safety margins, in the context of the power uprate, would be of interest to the Austrian side. Clarification of apparently contradictory statements about the reduction of margins would be desirable, as well as detailed information on which margins are reduced to which extent.

A comprehensive discussion of all systems and components which could be concerned by a power uprate would also be of interest, including a discussion of all modifications implemented and planned.

The two phases of fuel development also are of interest to the Austrian side and should be explained in more detail; particularly concerning the schedule of the second phase, the burn-up which will be achieved then and its possible effect on source terms for DBAs and BDBAs.

The claim expressed in the **Answers** that the decay heat of the core will not increase proportionally to the power uprate should be further explained and supported both for the first and second phases of fuel development.

The planned schedule for implementation of the power uprate should be discussed in more detail, particularly regarding the question to which extent it permits collection and feedback of operating experience.

Of particularly great interest to the Austrian side are all questions concerning the potential effects of the power uprate in case of severe accidents – mainly, the reduction of intervention times and changes in the source term. As an important first step, provision of the Safety Report would be helpful for the clarification of the open questions.

Spent Fuel Storage

Spent fuel storage was not discussed in the **Austrian Statement**. Information on the present stage of development of the storage facility was provided in the **EIS**.

At Paks NPP, the concept of modular vault dry storage is employed, and not the cask storage concept which is employed in many other countries.

Assessment

The fuel storage facility at the Paks site already contains a large amount of radioactive materials, which will grow considerably in the coming decades if lifetime extension is implemented.

The storage concept employed appears to be more vulnerable to external impacts and terror attacks than the cask storage concept. Furthermore, it is likely that it will pose more problems in case of contamination of the store through a reactor accident.

Therefore, further information regarding the storage concept's vulnerabilities, and the possibilities of large releases from the store, would be of interest from the Austrian point of view.

There are indications that the planned storage duration (50 years) is likely to be exceeded. A discussion of this point, in the context of the planning for a final repository in Hungary, would also be of interest from the Austrian point of view.

Part I Technical Issues

Introduction

A number of Technical Issues have been selected for discussion in this report. They were selected according to their relevance for Austria, because of their direct connection to the question of severe (beyond design basis) accidents with potential cross-border releases, and/or because of their general importance in the context of lifetime extension and power uprating. With one exception, all of them have already been discussed in the Austrian Statement of September 2005.

Those issues are listed below:

1. Ageing Management Program
2. Reactor Pressure Vessel
3. Steam Generators
4. Confinement System
5. Seismic Hazards
6. Terror Attacks
7. Power Uprating
8. Spent Fuel Storage

For each Technical Issue, apart from Issue No. 8, the discussion is structured in the following manner:

- Introduction (explanation of the relevance of the Issue)
- Treatment of the Issue in the Preliminary Environmental Study and the Austrian Statement of September 2005
- Treatment of the Issue in the Environmental Impact Study and the Answers provided by Paks NPP
- Discussion of Treatment of Issues in EIS and Answers
- Information Provided by the Hungarian Representatives at the Hearing June 6, 2006
- Assessment (focusing on questions where further information would be of interest)

Issue No. 8 (Spent Fuel Storage) was not discussed in the Austrian Statement of September 2005, and accordingly, also not treated in the Answers provided by Paks NPP. In this case, the second and third parts as listed above have been merged into one (Treatment of the Issue in the Preliminary Environmental Study and the Environmental Impact Study).

The questions where further information would be of interest from the Austrian point of view are summarized for all Technical Issues in a concluding section.

TI Ageing Management Program

1. Introduction

All systems, structures and components are subject to ageing. Apart from the particularly crucial ones which are discussed separately, this concerns the pressurizer, the primary coolant pumps and the pipes and valves of the primary cooling circuit, components and pipes of the secondary circuit, as well as a multitude of SSCs with comparatively less safety significance.

Even failures and damages in systems, structures and components of lesser relevance for safety can be relevant for the overall plant risk; and plant risk will increase if such failures and damages become more frequent through ageing. “Small” failures can be precursors to more serious incidents, and the more often they occur, the higher the probability that one of them will indeed develop into an accident sequence, or increase the severity of an accident sequence not initiated by ageing.

Therefore, an all-embracing system of ageing management is required for an NPP, particularly in case of life extension. Ageing management is the totality of all administrative and engineering measures which are executed by the plant operator with the goal of controlling all ageing mechanisms relevant for safety, and of ensuring the availability of required safety functions throughout the plant's service life. The main task of ageing management consists of the recording of possible ageing mechanisms, and of the effective prevention of their adverse effects.

To a considerable extent, ageing management relies on and presupposes a functioning system of in-service-inspection – for example, ultrasound testing of the RPV and the primary piping, eddy current testing of steam generator tubes and other non-destructive tests.

2. Treatment of Ageing Management in the Preliminary Environmental Study and the Austrian Statement of September 2005

According to the **Preliminary Environmental Study**, there is an ageing management program implemented at Paks NPP which is being developed further as part of the planning for the lifetime extension. In section 1.2, there is mention of systematic monitoring of ageing, which was begun eight years ago, focusing on the reactor pressure vessel embrittlement, and erosion corrosion.

Furthermore, a program of registration of ageing effects, description of the changes they lead to, and determination of corrective action is mentioned, without presenting detailed information.

The results of the program concerning ageing effects, including brief indications which measures are required in case of a lifetime extension to 50 years are listed in section 3.2.2 and again, in a different context, in tables 6.1 and 6.2 of section 6. However, in this summary treatment, too, the system of ageing management is not described in detail.

Furthermore, the listing is restricted to building structures and mechanical components and systems (including emergency diesel generators, ventilation, off-gas treatment and waste water treatment). The whole complex of electrical and I&C-systems is summarily dealt with in one sentence. Areas like operating management systems and documentation are not discussed at all in this context.

In the **Austrian Statement on the Preliminary Environmental Study** it is emphasized that ageing in an NPP, even regarding SSCs of comparatively lesser safety significance, is of importance for the risk of extended plant operation. This issue is therefore connected to the issue of severe accidents with possible consequences for the Austrian population.

Therefore, it was stated that the ageing management program and questions associated with it need to be dealt with in more detail in the further course of the environmental impact assessment of the lifetime extension of Paks NPP, including the presentation of past experiences with ageing, in particular regarding incidents which have occurred because of ageing effects.

3. Treatment of Ageing Management in the Environmental Impact Study and the Answers provided by Paks NPP

In the **EIS**, the information already provided in the Preliminary Environmental Study is repeated in section 3.2.2 and tables 6.1 and 6.2, with very few additions which reflect the latest state (2005).

The **Answers** provide somewhat more information on the ageing management program of Paks NPP. It is pointed out that systematic ageing management activities were introduced about ten years ago. Those activities are performed in addition and support to the Periodic Safety Reviews (planned in ten-year intervals) which were introduced in Hungary in 1993.

The systematic ageing management system reportedly has been established and developed on the basis of several regulatory body's guidelines as well as recommendations of the IAEA.

All documents and information relating to ageing management of important equipment (as identified according to the regulator's guidelines) are available in a separate display system and database established for monitoring ageing management (DACAAM system).

Apart from the critical (non-replaceable) components, the status of other structures, equipment and components is also controlled as part of ageing management.

In the framework of the licensing procedure for lifetime extension, the ageing management program for safety-related passive components is to be reviewed by requirement of the licensing body. This review is performed according to the methods applied by U.S.NRC in the course of license renewal, considering ten main steps which are being listed in the Answers.

Ageing management of the large number of active components is being monitored by the maintenance effectiveness monitoring system, which is currently being introduced.

4. Discussion of Treatment of Ageing Management in EIS and Answers

The information on ageing management as presented in the EIS and, particularly, in the Answers makes clear that the system is not yet completed, particularly not in the context of lifetime extension. Parts of the system, on the other hand, appear to be well implemented and, for some years, successfully performing.

It has to be kept in mind, however, that the regulatory system in Hungary is in the process of being changed, regarding in-service-inspections, which constitute the basis for ageing management. The main incentive for introducing this new approach is *"[r]educing inspection efforts without safety level decreasing"* [CSNI 2005]. This includes plans by the licensee to reduce ISI frequency for safety-relevant equipment (see also Technical Issue Steam Generators). New approaches will have to be developed during the next years, for example:

- Determining extent of inspections by risk ranking;
- introducing quality criteria for probabilistic risk analyses which shall be used for developing risk-informed ISI programs;
- verification and validation of fracture mechanics codes and structural reliability models.

It will be necessary to follow those new trends and approaches as the licensing procedure for lifetime extension proceeds; information should be available on their development.

This is particularly important since regarding IAEA Safety Standards relevant for ageing management, which are mentioned as important for the Hungarian approach in the Answers, the situation is also in flow, and rapidly evolving. At the moment, a fairly large volume of guidance documents is available from the IAEA concerning ageing management; but so far, there are no higher level guidance documents which identify the key elements of effective ageing management, and show how they fit together. Therefore, the IAEA is preparing a Safety Guide "Ageing Management for Nuclear Power Plants and Research Reactors", at present in the draft stage, and is planning a Safety Standard document on "Safety Aspects of Ageing Management" [IAEA 2006].

5. Information Provided by the Hungarian Representatives at the Hearing June 6, 2006

The Ageing Management system at Paks NPP and its development were presented in a summary contribution by the Hungarian side.

The Hungarian system of regulations was briefly summarized. It was reported that in the context of the lifetime extension, the current Ageing Management program is being reassessed, applying 10 criteria as required and defined by U.S.-regulations. This reassessment is almost concluded; according to the report at the Hearing, it mostly led to a confirmation of the existing system, with only a small number of modifications required.

The DACAAM system for monitoring ageing management which already was briefly mentioned in the **Answers** was again discussed; it was reported that this system, developed in Hungary, has recently been acquired by the operators of the Finnish NPP Loviisa.

The information was provided orally; no written version of the presentation or other documentation was handed over at the **Hearing**.

6. Assessment

Considerable changes and developments are to be expected in the AMP program of Paks NPP during the next years. Further information on this process would be of great interest from the Austrian point of view. As an important first step, provision of the information reported orally at the Hearing in written form would be helpful.

In particular, further observation of this issue should permit to ascertain that the new approach to in-service-inspections to be introduced at Paks, which is to include reductions in inspection efforts without a decrease of the safety level, indeed does not lead to any safety level decreases.

TI Reactor Pressure Vessel

1. Introduction

The reactor pressure vessel is the central component of a nuclear power plant. It contains the reactor core, consisting of nuclear fuel, where the heat production through a nuclear chain reaction takes place. During operation, the reactor pressure vessel is subject to intense neutron irradiation, as well as high temperature and pressure.

The most important ageing mechanism of the reactor pressure vessel is embrittlement of materials close to the core through neutron irradiation. Embrittlement stands for reduction of toughness as well as a shift of the ductile-to-brittle-transition temperature T_k to higher values – meaning that the material is still in a brittle state, and hence more prone to brittle failure, for increasingly higher temperatures. Impurities like copper and phosphorus favour embrittlement, as well as nickel and manganese. The importance of embrittlement is high for VVER reactors due to the high neutron fluences encountered at their vessels.

The embrittlement of the reactor pressure vessel increases the hazard of vessel bursting – particularly in case of the injection of emergency core cooling water during an incident, which leads to cooling of the vessel wall (so-called thermo-shock). The failure of the pressure vessel constitutes a beyond design basis accident for all light water reactors. Furthermore, pressure vessel failure can lead to immediate confinement (or containment) failure as well, for example through the pressure peak after vessel bursting. A core melt accident with high and early radioactive releases would be the consequence.

2. Treatment of the RPV Issue in the Preliminary Environmental Study and the Austrian Statement of September 2005

The embrittlement of the reactor pressure vessels is discussed in section 3.2.2 of the **Preliminary Environmental Study**. The critical ductile-to-brittle-transition temperature is given as 140°C. This value was determined in calculations for the Finnish NPP with a VVER 440/213 (Loviisa). The analyses on which it is based are not described. Furthermore, no justification is provided that this result also applies for Paks NPP.

The expected values for the ductile-to-brittle-transition temperature for up to 50 years of operation are presented in a table. One value for base material and welds each is given per unit (presumably, the maximum value reached), without specifying the location. There is no description of the surveillance program and no explanation how those values were determined.

It is pointed out that the welds of units 1 and 2 come close to the critical temperature and that annealing might become necessary (again, the Finnish experience is mentioned in this context).

In order to mitigate possible thermo-shocks, the water in the tanks of the emergency core cooling system is to be heated at units 1 and 2, beginning in the 24^{th} year of operation (i.e. 2006 and 2008, respectively). Similar measures which were implemented in Finland are briefly described.

In section 6, table 6.2, of the Preliminary Environmental Study the consequences of embrittlement are summarized by stating that only a minor increase of the accident risk is to be expected. This statement is not quantified or discussed.

In the **Austrian Statement**, it was listed what has to be available, in order to contain the hazard of reactor pressure vessel bursting, regarding various material data, in-service-inspection, thermo-hydraulic and fracture mechanics analyses and counter-measures.

It was pointed out that although there was no reason to assume that the items listed were not available for Paks NPP, the Preliminary Environmental Study did not provide information concerning most of those items and information was provided in a very summary manner only. In conclusion, it was stated that the issue of pressure vessel embrittlement would have to be presented and discussed in a comprehensive, detailed matter in order to permit a meaningful assessment.

3. Treatment of the RPV Issue in the Environmental Impact Study and the Answers provided by Paks NPP

In the **EIS**, section 3.2.2, various systems and components of the Paks units are discussed, regarding their state and the conditions for life extension. The reactor pressure vessel is also treated there.

Again, the discussion of RPV embrittlement and resulting safety questions is based on the critical value T_k = 140° C as determined for the Loviisa NPP. In Finland, annealing of the RPV of unit 1 was performed with a safety margin of 4°, when T_k = 136° C had been reached. In Paks, this value will not be approached, according to EIS, for the base material. However, the weld material of unit 1 and 2 are forecast to reach 136° C and 128° C, respectively, after 50 years of operation. Thus, annealing might become necessary.

Furthermore, again analogous to measures which had been taken in Finland, the water temperature of the hydro-accumulators and of the high pressure emergency core cooling system is to be increased after 24 years of operation. The purpose of this measure is to reduce the loads arising from thermo-shock in case of an accident. Also, the discharge head of the high pressure system is to be reduced.

By changing the core configuration (low-leakage-core), neutron irradiation of the vessel wall has been reduced by at least 30 %.

It is pointed out that the RPV is being inspected partly every year, and in its entirety in four-year intervals. The progress of embrittlement is constantly monitored with surveillance samples.

In the course of the licensing procedure for the lifetime extension, the values forecast for the ductile-to-brittle transition temperature will be re-evaluated. It is expected that it will be possible to reduce it, leading to a reduction of the probability that annealing will become necessary. Furthermore, analyses of pressurized thermal shock (PTS) scenarios will be performed which will permit a re-evaluation and revised planning for counter-measures like increasing cooling water temperatures.

In the **Answers** provided by Paks NPP, the ageing management of the RPV is briefly described. It is emphasized that a complete new safety and component ageing analysis is required for licensing the lifetime extension. A full manufacturing database, a surveillance program, a material ageing database, periodic non-destructive testing covering all relevant parts, as well as a complete new set of PTS analyses have to be demonstrated in this context. (The points mentioned correspond mostly – but not completely - to the listing in the Austrian Statement on the Preliminary Environmental Study.)

The thermo-hydraulic transients constituting the basis of those analyses have already been identified and modeled. The PTS analyses are to be performed in accordance with the appropriate Regulatory Body's guide, which is taking into account IAEA recommendations and the VERLIFE program.

On the basis of analyses existing so far, it is claimed that no obstacles for extending the lifetime by 20 years have been identified. Adequate safety measures are to be taken if required according to the new analyses.

Brief information on the original status of the reactor pressure vessels, as well as on the surveillance programs, is also given. It is claimed that adequate data are recorded for the original, un-irradiated condition. Regarding surveillance, it is stated that the same set of specimen is applied at Paks, as at Loviisa. Inside each reactor of Paks NPP, six original sets of specimen were placed. The neutron flux at the location of those specimens is almost 12-19 times larger than the one affecting the inner surface of the reactor wall. Hence, the specimens taken out after 4 years represent damage corresponding to an operating lifetime of 48-76 years for the vessel. The specimens and their testing residue are stored in a way they can be identified when required. The test results are reviewed by advisory experts, and recorded.

Regarding power uprating, it is recognized that this could lead to an increase of the neutron flux in the RPV vessel wall. However, it is stated that due to the application of a Hafnium cover in the upper part of the control assemblies and to applying low leakage schemes, the neutron fluence at the internal vessel surface will actually decrease (a similar, briefer statement is contained in the EIS).

4. Discussion of Treatment of RPV Issue in EIS and Answers

It is obvious from the presented information that the safety assessment of the Paks RPVs, in connection with the lifetime extensions, is yet in an early stage and no definite and final results are available; hence, no definite judgment is possible at this moment.

The values given for the expected ductile-to-brittle transition temperatures are subject to revision. The use of Loviisa results regarding the critical T_k value can only serve as rough orientation anyway, since no proof has been presented that those results also apply for Paks NPP. Even nuclear power plants of the same reactor type will usually differ in some design details.

Even taking the early stage of considerations into account, however, more details would have been desirable, for example regarding the following topics:

- Database for un-irradiated material – quality control applied, scope of the recording of properties
- Comprehensive description of surveillance sample program, including all sets
- Detailed description of the framework for PTS analyses, listing of thermo-hydraulic calculations already performed
- Methodology to be applied in the fracture mechanics part of the PTS analyses

Some important questions have not been addressed at all in the EIS or the Answers. This concerns particularly the safety margins which are to be applied in the PTS analyses. This point is only mentioned in passing, for Loviisa (where a margin of 4° between the critical T_k and the T_k actually reached was applied). A "[d]efinition and justification of an appropriate safety margin", as mentioned in the Austrian Statement on the Preliminary Environmental Study, was not provided.

It has to be noted that 4° represents a rather small margin. In the IAEA "Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants" [IAEA 1997a], for example, a safety factor of 10° is recommended. The IAEA Guidelines also state that other values could be used, if they are justified. However, a justification is lacking in the EIS and the Answers.

Regarding the fracture mechanics calculations, considerable differences exist between the IAEA and VERLIFE methodologies. These differences concern, for example, assumptions on crack depth, crack shape, integrity of RPV wall cladding and the application of the warm pre-stressing (WPS) effect [LAMPRECHT 2005]. The selection of methodology has to occur early in the course of the PTS-related work and can have significant influence on the results. Therefore, it appears appropriate that information should be provided at an early stage.

There is no mention in EIS and Answers of the dose rate effect which might affect the surveillance results. At low neutron fluxes, embrittlement may be accelerated and hence, the shift of the ductile-to-brittle transition temperature per unit dose can be higher than for high neutron fluxes. This effect is particularly problematic if embrittlement prediction is based on surveillance samples with fluxes which are considerably higher than the flux in the RPV wall (as is the case in Paks, where the so-called lead factor is in the order of 12 to 19).

In fact, an extended surveillance program has been implemented in Paks, in addition to the original program, in order to provide a more extensive database. The specimen sets for this extended surveillance include Charpy specimens located in low flux positions. These specimens are to be used to evaluate the dose rate effect [IAEA 2005].

In EIS and Answers, no distinction is made between original and extended surveillance program, and the dose rate effect is not discussed. This is remarkable since the potential importance of this effect appears to have been well recognized at Paks NPP, considering the fact that special specimens have been dedicated to its study.

Regarding power uprating, the reduction of neutron fluence due to new fuel elements and a low-leakage core configuration is further discussed in the section dealing with the Technical Issue Power Uprating.

Another point which has not been discussed in EIS and Answers are the possible consequences of changes in the emergency core cooling system, particularly the decrease of the injection head of the high-pressure part of this system, for the control of accidents. The analyses which are to be performed appear to concentrate solely on PTS, i. e. on the aspect of thermal shock to the RPV. It is not clear to which extent the consequences of the ECCS modifications regarding other aspects than RPV wall cooling – in particular, adequate core cooling – are to be investigated.

5. Information Provided by the Hungarian Representatives at the Hearing June 6, 2006

At the **Hearing**, the TI RPV was touched only very briefly and summarily. It was reported that the Hungarian guideline No. 317 regulates the RPV safety analysis, and that inspections are performed according to VERLIFE and/or ASME XI. No further details were provided.

6. Assessment

The safety assessment of the Paks RPVs, in connection with lifetime extension, is still in an early stage. Results provided so far will be subject to revision.

Even at the present stage, some information should already be available which would be of great interest from the Austrian point of view. This concerns for example the database for un-irradiated materials, a comprehensive description of the surveillance program, information on the scope of the PTS analyses, the thermo-hydraulic calculations already performed, and the methodology to be applied regarding fracture mechanics.

Of particular interest is the question of the safety margin to be applied in the PTS analyses, as well as assumption on cracks, on RPV cladding integrity, and on the manner of WPS effect application. Furthermore, the possible effects of a dose rate effect and the way this effect is investigated in the surveillance program would be of great interest from the Austrian point of view, as well as a proof for the claimed reduction of neutron fluence in the vessel wall in spite of power uprating.

Possible consequences of changes in the emergency core cooling system for the course of accidents would also be of interest.

As the new PTS analyses will be performed in the next years, further information on methodology and results would be of great interest to the Austrian side.

TI Steam Generators

1. Introduction

In pressurized water reactors including VVERs, steam generators provide the link between the primary and secondary cooling circuit. The heat produced in the reactor is transferred to the steam line leading to the turbine. Steam generators are designed for optimal heat transfer between the two circuits, while providing a leak-tight boundary between them.

Steam generator ageing is particularly hazardous if it weakens the separating border between primary and secondary circuit (the wall of the steam generator tubes). A leakage between the two circuits implies a loss of coolant which is bypassing the containment.

Hence, the cooling water lost is not available for the emergency core cooling system. Furthermore, there is a direct pathway for releases into the atmosphere, potentially leading to large source terms.

Ageing processes can also concern other parts of the steam generator, possibly reducing the capacity of this component to draw off heat from the primary circuit.

Corrosive and erosive damage in steam generators has occurred repeatedly world-wide, as well as wall thinning. These problems have led to comprehensive ageing management activities. Increasingly, they include exchanges of the whole components.

2. Treatment of the Steam Generator Ageing in the Preliminary Environmental Study and the Austrian Statement of September 2005

Possible ageing problems of steam generators are discussed very briefly in section 3.2.2 of the **Preliminary Environmental Study**. It is mentioned that in-service-inspection takes place every four years and that stress corrosion can occur both at the primary and the secondary side (in spite of changes in the secondary water chemistry which have already been implemented).

Countermeasures envisaged are the exchange of the collector (regarding primary-side corrosion and erosion) and plugging of steam generator tubes (regarding secondary-side stress corrosion). It is pointed out that an exchange of the whole steam generator usually is regarded as necessary only if more than 20% of tubes have to be plugged. It is considered unlikely that an exchange will be required during 50 years of operation.

In section 6, table 6.2, it is again emphasized that it is unlikely for an exchange of steam generators to be required. It is also pointed out, however, that an exchange could be performed within 3 months, if necessary.

The severe incident in unit 2 in April 2003, occurring in the context of fuel element decontamination, is mentioned in section 5.5.3. However, the connection between fuel element contamination and corrosion problems in the steam generators is not discussed in the Preliminary Environmental Study.

In the **Austrian Statement on the Preliminary Environmental Study**, it was pointed out that corrosion of steam generators is an important issue for VVER plants and has created considerable problems at Paks NPP in the past – problems which apparently are not completely resolved yet.

Severe deposition of corrosion products (cruds) on the primary side of steam generator tubes has been reported to occur. Magnetite corrosion products (cruds) reached the reactor core, contaminating fuel element surfaces, and after partial remobilization, getting stuck at the lower spacer grid – leading to extensive fouling of fuel elements.

This not only led to the necessity of fuel element cleaning (and hence, the incident of April 2003 in unit 2), but also to a non-uniform distribution of the flow rate of cooling water through the core.

Research is still under way to fully understand primary circuit corrosion and crud behavior, particularly in the reactor core.

It was pointed out in the Austrian Statement that these issues have to be treated in the further course of the environmental impact assessment of the lifetime extension of Paks NPP. Furthermore, questions of in-service-inspections of steam generators require attention. The accuracy of the eddy current inspection methods, the scope, methods and results of the analyses for the determination of the remaining thickness of tube walls to guarantee integrity, questions concerning mechanisms and speed of crack growth and the plugging criteria derived on this basis have to be discussed.

Also, it was stated that the issue of steam generator exchange needs to be dealt with in more detail. In the Preliminary Environmental Study, an exchange is assessed as unlikely, but feasible. Another source quoted in the Austrian Statement, on the other hand, emphasized that steam generator replacement in Paks was not a realistic option due to high costs [DAVIES 2002]. It is not quite clear to which extent an exchange of the Paks steam generators is actually considered.

3. Treatment of the Steam Generator Ageing in the Environmental Impact Study and the Answers provided by Paks NPP

In the **EIS**, section 3.2.2, various systems and components of the Paks units regarding their state and the conditions for life extension are described. The steam generators are also treated there.

The information already provided in the Preliminary Environmental Study is repeated. In addition, a table giving an overview on tube plugging until 2005 is included. The highest percentage of plugged tubes is found in steam generator no. 3 of unit 2 (3,522 %). Most plugging rates are below 1 %.

It is reasoned that the plugging rate most likely will not reach proportions necessitating an exchange of steam generators. This is mainly due to two reasons:

- Modifications of secondary side water chemistry, in connection with the exchange of components in the secondary circuit, employing new structural materials less prone to corrosion/erosion;
- favorable chemical composition of the austenitic steel of the steam generator tubes.

The incident of April 2003 is discussed at some length in section 5.3.6 of the EIS. This discussion, however, is restricted to the incident as such, its direct consequences and the measures taken to avoid similar incidents in the future. The underlying problems leading to the necessity of cleaning of fuel elements – in particular, steam generator corrosion and fuel element fouling – are not addressed. The incident is further treated briefly in section 5.5.3 of the EIS, repeating the information already provided in the Preliminary Environmental Study.

The **Answers** provided by Paks NPP contain some summary information on the steam generators, concerning the design of the steam generators and the materials used.

It is pointed out that VVER-440 steam generators with stainless steel piping are not particularly sensitive to stress corrosion; as opposed to steam generator in other PWRs where the tubes are made of inconel.

It is pointed out that eddy current testing (ECT) is applied every four years to monitor wall thickness. In case of severe wall thickness reduction, the damaged pipe will be plugged. It is mentioned that severe criteria exist for plugging, in agreement with international practice; however, plugging criteria are not described in detail.

The feedwater distributing system in the steam generators has been replaced. The new material employed is not specified; it is likely that stabilized austenitic steel is now being used [IAEA 1997b]. Thus, erosion-corrosion processed ought to be significantly reduced. It is also mentioned that secondary chemistry was modified between 1997 and 2000, leading to further reduction of stress corrosion.

As in the EIS, it is emphasized that it is not expected that the necessity for steam generator replacement will arise.

4. Discussion of Treatment of Steam Generator Ageing in EIS and Answers

It is obvious from the presented information that the safety assessment of the Paks steam generators, in connection with the lifetime extensions, is yet in an early stage and no final results are available; hence, no definite judgment is possible at this moment. Indeed, it is pointed out in the Answers that Paks NPP is still in the preparation phase of lifetime extension and the analyses relevant for licensing are being elaborated now.

The position that tube plugging is not likely to lead to the requirement for steam generator replacement appears plausible considering that the IAEA does not regard VVER steam generator tubing as particularly troublesome. Relatively low plugging rates are reported [IAEA 1997b]. The percentages of plugged tubes as given in the EIS is roughly compatible with this observation, although some of the steam generators will require close further observation (for unit 2, in particular, all plugging rates are above 1 %, and rates are above 3 % in two cases).

In the EIS, it is mentioned that the usual limit for steam generator exchange lies at 20 %. No other limit is given. This is somewhat misleading since another source specifies, for Paks NPP, a maximum admissible plugging rate of about 15 %, according to a preliminary analysis [KATONA 2004]. The EIS also does not mention the crucial role of the Nickel content of the austenitic steel tubes. This content is in the range of 9 – 12 %; at Ni contents below 10 %, the susceptibility to outer diameter stress corrosion cracking is significantly increased [KATONA 2004]. Both these issues need to be discussed in more detail.

Erosion-corrosion of the feedwater distribution system is an important issue for VVER-440s. It can lead to the creation of free parts (feedwater distribution nozzles) which can cause damage to the steam generators or valves [IAEA 1997b]. It has been reported in the Answers that replacement of this system has occurred at Paks; however, due to the potential importance of this issue, more detailed information would be required (in particular, regarding the operating experience since the replacement has been performed).

Furthermore, questions of in-service-inspections of steam generators require attention. The accuracy of the eddy current inspection methods, the scope, methods and results of the analyses for the determination of the remaining thickness of tube walls to guarantee integrity, questions concerning mechanisms and speed of crack growth and the plugging criteria derived on this basis have to be discussed.

It was frequently stated that inspections occur in intervals of four years. In a recent report [CSNI 2005], it is pointed out that, in Hungary, the plant licensee is planning to modify the inspection frequency for some (not specified) safety relevant equipment from 4 years to 8 years, using risk-informed strategies to support this change in in-service-inspection strategy (see also the Technical Issue Ageing Management Program). Apparently, the regulator is still considering this modification. It should be clarified whether steam generators are also concerned by it; and if so, how reduction of the frequency of in-service-inspections is being justified.

The question whether steam generator replacement is considered as a realistic option, or if the unit concerned would rather be shut down, also has not been completely answered so far. It appears that this point is to remain open and will only be decided if and when a decision would actually be required.

Also of interest would be information on follow-up work on the incident of 2003, particularly regarding aspects relevant to corrosion problems, which again can be relevant for the lifetime extension.

5. Information Provided by the Hungarian Representatives at the Hearing June 6, 2006

It was reported that steam generator inspections originally were performed according to Soviet regulations. Several changes were implemented later; at the moment, implementation of procedures according to the U.S.-regulations ASME V and XI is planned by the operator, and in the course of being licensed.

Regarding qualification of inspection procedures, the European ENIQ system is at present being introduced.

The latest step of development of the inspection system will be completely implemented by 2008 – 2011 (this range of time apparently refers to stepwise implementation at the four units), if implementation of the ASME structure will be successful. In this case, there will also be some extension of inspection intervals (i. e., reduction of inspection frequency).

Regarding the scope of steam generator inspections, Soviet regulations required 10 % of steam generator tubes. This was later increased to 25 %, and recently to 100 %.

The limit for tube plugging was reported to be 10 %. It was emphasized that the value of 20 % given in the EIS referred to Western practice and that the practice at Paks was much more restrictive and conservative. Furthermore, it was reported that since 2000, there was a decreasing trend regarding the numbers of tubes plugged each year. This is to be due to a change in chemistry (higher pH-value in the secondary circuit).

The information was provided orally; no written version of the presentation or other documentation was handed over at the **Hearing**.

6. Assessment

The safety assessment of the Paks steam generators, in connection with lifetime extension, is still in an early stage. Also, as became clear at the Hearing, the development of the inspection system is still ongoing and will not be completed for several years.

Additional information regarding the role of the Nickel content in stress corrosion cracking of steam generator tubes could be of interest from the Austrian point of view. The same applies to the issue of erosion-corrosion of the feedwater system and the measures taken in this context.

Information on in-service-inspections, including discussion of the ongoing development of the practice in the next years, also would be of interest, as well as information on the follow-up work on the accident in 2003, particularly regarding aspects relevant to corrosion problems.

The provision of the information reported orally at the Hearing could be an important first step in clarifying those issues.

TI Confinement System

1. Introduction

The confinement system of VVER 440/213-plants consists of a system of rooms, containing the primary circuit, with a steel liner to minimize leakages, the barbotage tower with large trays filled with water and air trap (for passive pressure suppression by condensation of steam in case of accidents) and an active spray system.

The behavior of the confinement system is of crucial importance for all severe accidents where the confinement is not damaged at an early stage by a massive impact, or by-passed. In such cases, the time and extent of radioactive releases is determined by the ability of the confinement to withstand loads more severe than the design basis, and its leak-tightness.

Furthermore, regarding design basis accidents, failure of the confinement in this case can lead to a transition into a beyond-design-basis accident, with an increase of radioactive releases and possibly an aggravation of the accident sequence, due to the loss of cooling water out of the confinement, which is lost to the sump of the reactor building and hence, for later emergency core cooling.

2. Treatment of the Confinement Issue in the Preliminary Environmental Study and the Austrian Statement of September 2005

Ageing problems of the confinement system are discussed briefly in the **Preliminary Environmental Study** (section 3.2.2 and also in section 6, table 6.2.) With one exception, it is regarded as sufficient if the maintenance, repair and exchange work as practiced for the operating period of 30 years is extended for the additional two decades. The only recommendation beyond that is to exchange the seal bushings of the confinement spray system.

In the **Austrian Statement on the Preliminary Environmental Study** it was suggested that potential ageing problems of this system need to be dealt with in more detail in the further course of the environmental impact assessment of the lifetime extension of Paks NPP.

In particular, it was emphasized that the long-term behavior of the steel liner of the confinement rooms and possible implications for confinement leak-tightness would deserve attention. Furthermore potential ageing effects to the barbotage condenser system which would be subject to considerable loads during accidents, and the safety reserves which can be of importance in case of beyond-design-basis accidents.

3. Treatment of the Confinement Issue in the Environmental Impact Study and the Answers provided by Paks NPP

In the **Environmental Impact Study**, essentially the same information concerning confinement ageing as in the Preliminary Environmental Study was provided, with an addition concerning the renewal of the isolation of the roof to take place 2005/06.

Furthermore, the resulting confinement pressures for various design basis accidents are presented in section 5.5.2.3.2. It is reported that the actual values remain well below the maximum design overpressure in the confinement, as well as the maximum pressure difference acting on the barbotage condensers.

In the **Answers** provided by Paks NPP it is summarized that in case of a possible accident, the safety level of the Paks NPP is maintained and the environment is protected by safety and localization systems consisting of active and passive components. The passive protection functions are provided by the intended parts and structures of constructions. Beyond general structural stress such structures were designed to meet the following requirements: Isolation, protection against overpressure, protection against inner splinters, radiation protection.

It is reported that the construction is monitored and inspected during the operation on the basis of status control and maintenance (ageing management) programs. Types and extent of the ageing and deterioration processes experienced and expected correspond to the international experiences.

The regular main reviews including inspection of the coating of the confinements and bubble condenser towers and steel sheet covers are listed.

Possible corrosion phenomena detected during inspections are eliminated and failures are repaired on the basis of detailed technological procedures that include proved and practically tested methods. It is stated that no deterioration of reinforced concrete and concrete steel structures has been experienced so far.

Measures to eliminate leakages detected during the operation so far are reported (repair of roof insulation, elimination of leakages, modification of water draining when technological systems are discharged, repair of dilatation elements, etc.).

About the state of the containment and the main building, it was stated that following the ageing management, status control and maintenance programs used at the plant the conditions of long-term and safe operation are ensured. On the basis of the reviews no nuclear safety-related deficiencies have been detected.

In the frame of the preparation procedure of the lifetime extension licensing the overall review of the ageing management programs relating to constructional components is now in process.

4. Discussion of Treatment of Confinement Issue in EIS and Answers

The treatment of the confinement system, which is of great relevance for plant safety, is still rather brief and summarily.

Particularly, there are no comments on ageing of the barbotage condenser system, and the safety reserves of the whole confinement system which would be relevant in case of a beyond design basis accident.

The bubble condenser confinement system is a unique Soviet design. In the 1990s, tests were conducted with EU support (as a PHARE project) which demonstrated that the bubbler condenser is capable to withstand the loads and maintain its functionality after a large break LOCA [WENRA 2000]. Further experiments were performed by the utilities, and work to resolve some remaining questions was carried out in 2002, effectively confirming the earlier results [CSNI 2003].

The general conclusions drawn from the experiments were as follows:

- The test parameters measured by different transducers provide values that are generally in agreement within the error bounds.
- The discrepancies between the measured and calculated values are not significant and the character of the predictions is conservative.
- The observed differences between the measured and calculated values can be explained.
- The maximum pressure experienced in the tests is far from the design pressure of the containment system (0.25 MPa).
- The maximum pressure load on the tray walls measured during the tests, is far less than the 30 kPa limit value.
- Water level fluctuations were experienced but were found to be minor and disappeared when the steam started to flow into the bubbler condenser pool.
- Condensation-oscillations were not found within the investigated experimental conditions.
- The sequences investigated in the tests do not cause any significant challenge for the VVER-440/213 type bubble condenser and localization system.

The results of accident analyses presented in 5.5.2.3.2 of the EIS appear well compatible with those conclusions.

However, these investigations and tests concerned solely design basis accidents. Regarding protection against severe accidents (BDBAs), it is noteworthy that containment capability to limit releases appears to be somewhat inferior to Western PWR containments [WENRA 2000].

There is no discussion of this point in the EIS or the Answers. Presumably, the confinement system has some limited mitigative capabilities in case of a BDBA, even if it is inferior to that of Western PWR containments.

It was reported in 1997 that there are critical components of the bubble condenser which may fail under relatively low pressure difference load. There were consideration to strengthen certain parts of the condenser (inverted u-cap and I-beams), because the loss of their structural integrity could cause loss of condenser functionality, and subsequently an over-pressurization of the confinement after LOCA. It was also pointed out that the final decision regarding backfitting was to be taken after the experimental investigations and tests mentioned above were performed [KATONA 1997a].

No backfitting measures to the confinement are mentioned in the EIS (for example, in section 2.1.6, where the most important recent backfitting measures are listed). There is no discussion of the experimental results and their consequences regarding backfitting measures.

The confinement systems of VVER 440/213 plants have been designed with relatively high leak rates, compared to Western PWRs. For Paks NPP, a design value of 14.75 % at peak design pressure (0.25 MPa), and an actual leak rate as confirmed by tests of 9 % (with 2 % accuracy) was reported in the late 1990s [KATONA 1997b]. Typical leak of Western PWRs with full pressure would be around 1 % and lower.

This leak rate is of considerable importance regarding timing and extent of releases in case of a BDBA. There is no information in the EIS or the Answers concerning possible backfitting of the confinement system in order to reduce the leak rate.

5. Information Provided by the Hungarian Representatives at the Hearing June 6, 2006

The current confinement leak rates were reported to be in the range of one third to one half of the design leak rate (14,75 % per day), i.e. about 5 % to 7 %. (This indicates that leak rates are different for different units at Paks.)

It was stated that a comparison of the leak rates of the confinement system with leak rates of large dry containments of Western PWRs is somewhat misleading. Because of the pressure suppression system which is part of the WWER 440/213 confinement system, there will be high pressure for brief periods of time only during a design basis accident; high-pressure periods during DBAs can be considerably longer for PWRs with large, dry containments.

Regarding hydrogen recombiners, information contained in the EIS was repeated and it was emphasized that recombiners for hydrogen control in case of BDBAs might be installed. However, this issue still appears to be under examination and it has not been decided yet if the installation will take place, and which system will be used in this case.

The information was provided orally; no written version of the presentation was handed over at the **Hearing**.

6. Assessment

Because of the importance of the confinement system for plant safety, more detailed comments on ageing of the barbotage condenser system as well as on backfitting measures which were performed in the last years or are planned for the immediate future (particularly in connection to the PHARE experiments and investigations) would be of interest to the Austrian side – including the backfitting measures which have led to a reduction of the leak rate from 9 % (as reported 1997) to 5 – 7 %. The provision of the information reported orally at the Hearing could be an important first step in clarifying those issues.

Regarding DBAs, the time-pressure curves for the various cases considered would be of interest.

Among the DBAs analysed (including confinement behavior) and presented in the EIS (table 5.5.10), there are two for which the maximum configuration of the emergency core cooling system has been assumed (nos. 2 and 8; in the other 12 cases, there was a minimum ECCS configuration). The reason behind this assumption would be of interest, as well as the time-pressure curves for cases 2 and 8 with minimum ECCS configuration.

Also, a discussion of the behavior of the confinement system in case of a BDBA, including a discussion of its general functional capability in this case, as well as of safety reserves and of capabilities for accident mitigation, would be of great interest. All considerations in this respect should be based on the uprated power level (108 %) since the power uprate will be completely implemented at all units within a few years.

This discussion should include information concerning the consequences of the relatively high leak rates in case of BDBAs, of the possible duration of high-pressure periods during BDBAs, and of possible consequences of the leak rate for the timing and extent of releases in case of a BDBA.

As far as can be concluded from the available literature and the information provided (including the Hearing of June 6, 2006), there have been no systematic and comprehensive tests and investigations (comparable to those performed for DBAs) into the capabilities of the confinement in case of beyond design basis events up to date. Confirmation of this point, or information on investigations which have been performed nevertheless, would be of interest.

The further development regarding the possible installation of hydrogen recombiners for BDBAs would also be of interest from the Austrian viewpoint. This includes the discussion of alternatives as well as of possible disadvantages of recombiner use (higher temperatures in the confinement during BDBAs).

TI Seismic Hazards

1. Introduction

Earthquakes can lead to severe damage of a nuclear power plant, if the plant is not designed against the seismic loads which occur. A core melt accident can result, possibly with damage to the containment and large early releases. World-wide, seismic events are regarded as an important potential contributor to NPP risk. At many sites, they are the most significant risk factor among all external events and influences.

The assessment of seismic risks for an NPP is complex, and beset with many uncertainties. Basically, it consists of two steps: The evaluation of site seismicity, i. e. the determination of the maximum accelerations which have to be assumed at the site, for the design-basis earthquake (which belongs to the DBAs); and the evaluation of the seismic design of the NPP, i. e. the determination whether the buildings, structures and components of the NPP can indeed withstand this design-basis earthquake.

The basis for the evaluation of site seismicity and seismic design is continuously evolving; a process which has become particularly dynamic during the last decade.

Regarding site seismicity, new methods for geologic investigations have been developed in the last years. Furthermore, experiences from recent earthquakes have provided new insights [WENZEL 2004a].

Accordingly, the International Atomic Energy Agency published a new Safety Guide concerning “Evaluation of Seismic Hazards for Nuclear Power Plants” in December 2002 [IAEA 2002]. Also, the IAEA recently stated that their existing nuclear safety standards, which concentrated on new NPPs in the licensing phase, were not adequate for handling specific issues in the seismic evaluation of operating NPPs, and that a dedicated document was necessary. A Safety Report on “Seismic Evaluation of Existing Nuclear Power Plants” was published [IAEA 2003a].

Evaluation of seismic design also has made considerable progress in the last years, mainly based on the experience and measurements from recent seismic events. It became clear that traditional approaches do not satisfy the requirements of a realistic assessment of the seismic capacity of structures [WENZEL 2004b]. This development is mirrored by the publication of a new Safety Guide on “Seismic Design and Qualification for Nuclear Power Plants” by the IAEA [IAEA 2003b].

Regarding other problems related to site geology, instability of the ground can be an important issue, potentially leading to damages similar to those resulting from seismic events.

<u>2. Treatment of the Seismic Hazard Issue in the Preliminary Environmental Study and the Austrian Statement of September 2005</u>

In the **Preliminary Environmental Study**, questions of site seismicity are dealt with in section 4.3.4.1 and in appendix 3. For the determination of the design basis earthquake (in particular, the maximum accelerations to be assumed), IAEA guidelines were followed. The IAEA regulations played a particularly important role since specific national regulations were not available before December 1996 (and the new national regulations again were largely based on IAEA guidelines).

The maximum horizontal acceleration assumed for Paks NPP is 0.25 g, the maximum vertical acceleration 0.2 g.

The investigations which constitute the basis for those assumptions were mostly concluded by 1996. Some additional work was finished 1998, and 2000.

Regarding seismic design, it was briefly mentioned in section 2.1.6 that seismic backfitting of building structures and safety systems took place, without any specification. There was no systematic discussion of seismic design issues in the Preliminary Environmental Study.

In section 3.2.1, there was mention of instability of the ground around unit 4 of the Paks NPP, which can lead to subsidence of the soil and hence, to damages to buildings. It was noted that stabilization of the ground through injections might already become necessary during the first 30 years of operation.

The **Austrian Statement on the Preliminary Environmental Study** emphasized the importance of seismic events, which can lead to severe accidents with large releases and hence, with potential consequences for the Austrian population. It was pointed out that since this field was under rapid development in the last years, with new insights and experiences gained as well as methods developed, it is of particular importance that all investigations and analyses correspond to the most recent state-of-the-art.

However, the state-of-the-art as represented in the latest IAEA publications apparently had not been taken into account in the Preliminary Environmental Study and the investigations which are described therein. According to the Austrian Statement, it appears that the determination of site seismicity was based on the level of knowledge of the early 1990s.

Regarding seismic design, the Austrian Statement quoted other sources which stated that seismic upgrading has taken place in the late 1990s and the early 2000s. Due to the lack of information in the Preliminary Environmental Study, however, it could not be determined to which extent new information and new methods have been applied in this field recently.

It was concluded in the Austrian Statement that in the further course of the environmental impact assessment of the lifetime extension of Paks NPP, the issue of seismic hazards (including both site seismicity and seismic design) would have to be presented and discussed in a comprehensive, detailed manner. This should permit the assessment to which extent appropriate, state-of-the art data and methods have been applied, which additional work of seismic hazard re-assessment might be needed, the schedule of this additional work, and, eventually, its results.

The issue of ground instability (subsidence) concerning unit 4 should also be presented and discussed in detail in the further course of the environmental impact assessment, in particular regarding counter-measures which might be required up to the end of an extended lifetime (and beyond).

3. Treatment of the Seismic Hazard Issue in the Environmental Impact Assessment and the Answers provided by Paks NPP

In the **EIS**, the information already provided in the Preliminary Environmental Study is repeated in section 4.3.4.1 and appendix 6 concerning site seismicity, section 2.1.6 regarding seismic design, and 3.2.1 regarding ground instability.

In section 5.5.2.2 of the EIS, results of probabilistic safety analyses are presented and discussed which were not included in the Preliminary Environmental Study. Regarding the general importance of these results in the context of severe accident, this issue is discussed in Part II.

As the only external factor which can potentially lead to severe core damage, earthquakes were investigated in those probabilistic analyses. According to the results presented in the EIS, this factor is the dominating contributor to the core damage frequency: The overall CDF is given as 3.0×10^{-4} per year and unit; 86 % of this value are due to earthquakes.

The EIS states that measures for risk reduction concentrate on seismic backfitting. It is reported that measures are already under way, like strengthening of the scaffolding in the turbine hall as well as of bolted connections in the reactor hall. These measures, it is stated, will lead to a significant reduction of the seismic risk. This reduction is not quantified in the EIS.

The **Answers** provided by Paks NPP describe, in some detail, the activities on the assessment of seismic hazards until today. The investigations which were performed in the mid-90s, after IAEA experts pointed out inadequacies in the original site investigations and input parameters, are described. Three projects were carried out 1993 – 1995:

- Design and implementation of a local seismic monitoring network with systematic data collection and analysis;
- New geological, geophysical and seismological investigations in order to check and complete the available data;
- Geotechnical investigations to determine geodynamic properties.

As a result, the value of the maximum horizontal acceleration for the design basis earthquake (with a frequency of 10^{-4}/yr) was set at 0.25 g, as already reported in the Preliminary Environmental Study. A seismic reinforcement program was launched in 1998.

Furthermore, it is emphasized in the Answers that a new assessment of seismic hazards will be performed independently of the planned lifetime extension, in the framework of the next Periodic Safety Review (PSR) which is to be carried out for Paks NPP from 2006 to 2008.

During the PSR program, additional data acquisition and field surveys will be carried out corresponding to the recent state of the art. A new seismotectonic model will be constructed and seismic hazard will be reevaluated.

Preparation of the program started in 2005 by an integrated analysis of the results of the seismic monitoring network and construction of a 3D geologic-tectonic block model of the site area.

In the frame of the new seismic hazard assessment all the input data received from the assessment made in the mid-90s are to be reviewed. As far as required due to new developments, new measurements will be taken and the seismotectonic model will be constructed by this improved data system. Also, new probabilistic analyses will be performed for seismic scenarios.

4. Discussion of Treatment of Seismic Hazard Issue in EIS and Answers

It is appropriate and commendable that new investigations of seismic issues and a new assessment of seismic hazards – presumable employing the most developed, up-to-date methods – are planned to be performed in the next years, and indeed have already begun.

Furthermore, it is commendable that backfitting has already been implemented and is being continued, in order to increase resistance of buildings, systems and components to seismic events. It will also have to be noted whether the assessment of site seismicity will lead to further backfitting requirements.

Those developments which aim at a better characterization of the seismic risk at the Paks site, and at a reduction of this risk, notwithstanding, it must be emphasized that the value for the core damage frequency contribution of seismic events given in EIS (ca. 2.610^{-4}/yr) is high. It is significantly higher than the target value for CDF for existing nuclear power plants – 1 10^{-4}/year- which has been formulated by the International Nuclear Safety Advisory Group (INSAG) of the IAEA [INSAG 1999]. This high value was not quoted in the

Preliminary Environmental Study. In this study, estimates for the “total” core damage frequency in the order of 4 10^{-5}/yr were given for the Paks units in table 5.50; however, this “total” value did not include seismic events. The values from table 5.50 were also referred to in the Answers, without mentioning the seismic contribution.

This point requires more detailed presentation and discussion.

It is interesting to note that in another source [NSC 2005] the average value for the core damage frequency of a unit in Paks originating from a seismic event is given as 2.87 10^{-4}/yr, still somewhat higher than in the EIS, with a total CDF of 310^{-4} (as in the EIS). This discrepancy should be clarified, for example by providing and discussing the uncertainty bandwidths of the various CDF estimates.

This source [NSC 2005] also states that after all seismic backfitting measures have been completed, overall CDF would be reduced to 310^{-5}/yr.

With this situation in view, the following information would be of considerable interest in order to get a comprehensive and up-to-date picture of the seismic hazards associated with operation of Paks NPP:

- Detailed information on the seismic probabilistic analyses which have led to the high results;
- Information on the current state of the backfitting;
- Quantitative information on the reduction achieved by backfitting up to now, and to be achieved in the future.

Furthermore, as already pointed out, future activities in this field will have to be closely followed. Another aspect in the context of seismic hazards concerns the possibility of an abnormal flood being caused by earthquake damage to the Slovakian hydroelectric power plant Gabcikovo. A large amount of stored reservoir water could be released in this case. A discussion of this point would be of interest, including an analysis of possible consequences for the nuclear power plant, and, if applicable, of counter-measures.

5. Information Provided by the Hungarian Representatives at the Hearing June 6, 2006

It was reported that the value for the seismic contribution to the CDF in the EIS (which is also given in the final Safety Report for Paks NPP) was derived in a very conservative manner to begin with. Furthermore, an estimate was given for the reduction of CDF due to seismic backfitting implemented so far (among other measures, strengthening of the turbine hall): According to the results of new analyses (apparently from 2006), the seismic contribution has been reduced to about one quarter. The value of 6.6×10^{-5} per year has been quoted at the Hearing, corresponding to an overall CDF just below 10^{-4}/yr.

Furthermore, the Hungarian side provided summary information concerning the seismological investigations which were performed in the past years. For example, monitoring of microseismicity was mentioned, registering about 700 events in the past decade. The investigations provided the basis for determining the design ground acceleration (0.25 g, corresponding to an earthquake with a frequency of 10^{-4}/yr). This value is regarded as conservative by the Hungarian representatives. It was stated that Paks probably can be regarded, from a seismic point of view, as the safest nuclear power plant in Central Europe.

6. Assessment

Even with the high contribution to the core damage frequency due to seismic events as presented in the EIS being no longer valid, and a considerably lower number having been achieved according to the report at the Hearing, seismic events are still the dominant contributors to CDF at Paks.

More detailed information on the present state of the backfitting and the methodology and results of the latest seismic risk analyses would therefore be of considerable interest.

Further new investigations of seismic issues, including a new assessment of seismic hazards, will be performed in the next years. Because of the continuing importance of seismic issues, it will be of interest from the Austrian point of view to closely follow those investigations and assessments.

Further seismic backfitting activities which are planned at the moment or will be planned because of the results of future investigations, are of considerable interest and should be followed in the next years.

TI Terror Attacks

1. Introduction

It is general consensus that the topic of terror attacks should not be treated publicly in a manner which would provide "useful" information to terrorists and saboteurs, and/or provide them with new ideas for attack scenarios.

If this restriction is consistently taken into account, however, the issue of malicious human acts against NPPs can and should be discussed whenever NPP hazards (in particular, severe accident with possible cross-border effects) are dealt with – for the following reasons [HIRSCH 2005]:

- The terrorist threat appears to be particularly great in the 21st century.
- It is prudent to assume that a nuclear power plant can appear as an "attractive" target for terrorists – because of the potential long-term effects of radioactive contamination, the immediate effects on electricity generation and because of the symbolic character of nuclear power as typical "high-tech".
- Nuclear power plants are vulnerable to a broad spectrum of possible pathways of attack, including attack from the ground, the air, water ways, and by insiders; as well as to a broad spectrum of possible means of attack, including bombs, aircraft, shelling, missiles, application of explosives etc.
- An attack on a nuclear power plant can lead to radioactive releases equivalent to several times the release at Chernobyl.
- Certain protective measures against terror attacks are conceivable. However, they are not very effective.

These points apply to all types of commercial reactors at present being operated in the world. However, there are plant-specific differences, for example regarding vulnerability of spent fuel pools, robustness of the reactor building or spatial separation of other buildings and systems.

2. Treatment of the Terror Attacks Issue in the Preliminary Environmental Study and the Austrian Statement of September 2005

In the **Preliminary Environmental Study**, malicious acts of third parties against Paks NPP and their possible effects are not discussed at all.

In the **Austrian Statement on the Preliminary Environmental Study**, it was emphasized that a terror attack against Paks NPP could have consequences for the Austrian population. It was argued that because of the importance of this topic on the one hand, and the variations regarding vulnerability which give rise to the requirement of plant-specific analyses on the other hand, the issue of terror attacks and sabotage should be considered and discussed in the further course of the environmental impact assessment of the lifetime extension of Paks NPP, in order to obtain a better understanding of those consequences.

It was pointed out that, as far as it is known, the reactor buildings at Paks NPP are not designed against the crash of even a small airplane, implying a high vulnerability to terror attacks.

Regarding this issue, it was emphasized, the restriction regarding confidentiality as formulated above must be consistently and rigorously observed.

3. Treatment of the Terror Attacks Issue in the Environmental Impact Study and the Answers provided by Paks NPP

The **EIS** does not contain any discussion of the issue of terror attacks. Furthermore, external impacts like airplane crash and explosions are regarded to be very unlikely and hence are not considered either (the only external impact discussed in the EIS is earthquake). Quite apart from the fact that this could become a problem in the future because of increasing flight traffic in Central Europe, this generally implies a relatively low level of protection against any kind of external events, and hence a low level of protection against terror attacks.

The **Answers** provided by Paks NPP emphasize that the NPP meets the legal requirements concerning physical protection. No details are provided, but it is stated in summary that the international convention declared by the statutory law No. 8 of 1987, relating to physical protection of nuclear materials and facilities, the document of IAEA INFCIRC/225/Rev.4 and the relevant Hungarian laws and regulations (Act on Atomic Energy and BM (Hungarian minister of the interior) decree 47/1997. (VIII. 26.) modified by the BM decree 45/2005. (X.18) BM) are applied. Maintenance of the technical systems, training of the staff involved in physical protection and required developments are continuously ensured in order to maintain the level of physical protection. There are several developments in process and planned to be implemented at the plant before the beginning of lifetime extension, which will further strengthen the protection against terror attacks.

The level of protection of Hungarian nuclear facilities and relevant activities – also of the Paks NPP – is assessed every second year under the leadership of the authority (HAEA) in accordance with the decision made after the terror attack on 11 September 2001. The first assessment was implemented in 2002, followed by the second in 2004. During the assessments, terror threats as well as legal and preventive protection aspects of the country-wide preparedness are reviewed. Threats, physical protection and preparedness of the disaster management organizations for preventing the consequences of terror attacks were assessed in detail and recommendations were made for taking actions.

The main statement of the last assessment was that apart from the increase in general terror threats concerning the states of Europe, there was no indication that the risk factors had increased for nuclear facilities either internationally or in Hungary. Investigations concerning the protection of Hungarian nuclear facilities did not lead to any particular indication referring to the threat of terror attacks. The technical systems providing physical protection of the plant have been established, and they meet the relevant requirements, they are continuously maintained and technically developed. The operating and security guard staff is adequately qualified. The enforcement agencies involved in protection are in contact with both the plant and each other. The approved protection plans flexibly meet the actual situations.

Due to the high level of the above described physical protection and applied preventive protection, the Paks NPP does not seem to be an “attractive” target, according to the Answers. Robust construction of the primary circuit of the plant and the fact that high activity materials are stored in highly protected areas are claimed to disprove the possibility of large radioactive releases as described in the Austrian Statement.

4. Discussion of Treatment of Terror Attacks Issue in EIS and Answers

The Answers provide very general information which is not sufficient by far to disprove that large radioactive releases are possible after a terror attack. Indeed, these hazards exists for all commercial nuclear power plants; in addition, there appear to be some specific vulnerabilities at VVER 440/213 plants.

As concerns general protection against severe accidents that were not part of the original design basis, it is noteworthy that containment capability to limit releases is expected to be somewhat inferior to the Western PWR containments [WENRA 2000]. This can also increase vulnerability to terror attacks.

Other points which can increase this vulnerability are the lack of physical separation of redundant safety systems as well as shortcomings regarding fire protection identified for VVER 440/213 [IAEA 1999]; however, some backfitting took place in Paks which has brought improvement (i.e. rerouting of emergency feedwater and upgrade of fire protection) [WENRA 2000], such that these problems have been ameliorated.

An important weakness appears to be that there is no protection against crash of an aircraft at Paks NPP. This would also imply high vulnerability against other modes of attacks from the outside, for example shelling or application of explosives.

There is no explicit information regarding this point in the EIS or the Answers. However, EIS section 2.1.6 states that the upper part of the reactor building is build like any ordinary industrial building.

Furthermore, section 5.5.2.2 states that the probability of the crash of an aircraft onto the plant is so small that it this event need not be considered, which also indicates that design against crash of an aircraft was not regarded as necessary.

It must be emphasized that this topic can be discussed, if this is done in an appropriately general manner. Indeed, it has to be discussed. Since the consequences of a terror attack are potentially very high, and many people can be affected, people have a right to be informed about these risks. Furthermore, regarding protection against terror attacks, the public can actually be concerned by measures which are taken to increase security, even over national boundaries (e. g., regarding controls of flight passengers). This also gives rise to the need of appropriate information about the risks so that everybody can judge, to a degree, whether those measures are necessary and appropriate, can better understand the measures and last but not least, will be better prepared to cooperate.

To help deciding to which extent the topic can be discussed in public, the “Criterion of the Technically Competent Attacker Group” can be applied [HIRSCH 2005]: It does not appear problematic to openly discuss information which any group of attackers which is sufficiently competent to be able to plan and execute an attack with some likelihood of “success” possesses anyway, or can acquire with minimal research effort. Indeed, it would serve no purpose whatsoever to attempt to keep such information secret.

5. Assessment

The issue of terror attacks would be of great interest from the Austrian point of view, considering the large consequences such an attack can have, and the current increase of the general terrorist threat in Europe, acknowledged by the Answers.

Vulnerabilities, attack scenarios and potential consequences can and should be discussed in an appropriate general manner, and in an appropriate setting. Regarding public debates, the criterion should be applied that it would be pointless to attempt to keep information secret which a competent group of attackers can easily acquire anyway.

TI Power Uprating

1. Introduction

Increasing the electric capacity of a nuclear power plant beyond the original design value is generally referred to as power uprating. In principle, there are two ways to implement this goal:

- Increasing the thermal efficiency of the plant, at constant reactor power. This is achieved, in a PWR, through modifications in the secondary circuit.
- Increasing the thermal power of the reactor, generally by raising the coolant temperature. Thus, more steam is produced by the steam generators, and more electricity can be produced in the turbines (which will require modification).

In the first case (constant reactor power), plant safety remains practically unaffected.

In case of power uprating by increasing reactor power, the risk of plant operation can be increased. Margins relevant for safety might be reduced and plant ageing is accelerated.

One of the limiting factors for the raise of coolant temperature is the corrosion of the fuel element hulls, which grows more than proportionately with the temperature.

The radionuclide inventory in the reactor core is increased roughly proportionately to the power increase. A larger inventory implies a higher rate of decay heat, which accelerates the heat-up of the core in case of an accident and reduces the time until core uncovery.

The greater radionuclide inventory also has a direct impact in case of accidents since it implies increased releases. However, the inventory of long-lived radionuclides, which is particularly important in case of releases, depends on burn-up and hence is not necessarily increased with power uprating.

In order to assess the feasibility of a thermal power uprate, plant behavior during normal operation as well as during incidents must be considered. Among other things, the emergency core cooling system has to be examined, as well as the containment or confinement system.

Power uprate leads to an increase of the average maximum neutron flux on the inside of the reactor pressure vessel wall, if no counter-measures are taken. This increase can be of importance for pressure vessels with potential embrittlement problems.

2. Treatment of the Power Uprate Issue in the Preliminary Environmental Study and the Austrian Statement of September 2005

According to **Preliminary Environmental Study** section 2.1.6, the original capacity of 440 MW electric per unit has been increased, until 2003, to a nominal power of 472 MWe. The increases were achieved by improving the thermal efficiency; reactor power remained unchanged. In the introduction to section 6, it is mentioned that further power uprating is to take place within the next five to six years, to a nominal power of 500 MW electric. This is to be achieved by the use of a new type of fuel, modifications of the impellers of the main coolant pumps, and modifications in the secondary circuit. There is no further discussion or description of this envisaged uprating. Possible hazards arising from the reduction of safety margins caused by power uprating are not discussed in the Preliminary Environmental Study. The possibility of embrittlement acceleration due to increased neutron flux is not mentioned.

In the **Austrian Statement on the Preliminary Environmental Study** it was concluded that, as far as can be seen from the documents at hand, the power uprating to nominal 500We is dealt with – in the context of licensing procedure and environmental impact assessment – independently of the lifetime extension. Nevertheless, the effects of uprating have to be taken into account in the further course of the environmental impact assessment for the lifetime extension. Uprating and ageing can both potentially reduce safety margins; and interactions and synergistic effects between those two factors are possible (for example, regarding pressure vessel embrittlement). Furthermore, without the lifetime extension, power uprating would be hardly worthwhile, at least for the two older units.

Increased fuel corrosion and the acceleration of ageing processes have an indirect impact on the accident risk, and hence on potential consequences for the Austrian population.

The acceleration of accident sequences can have a direct impact on risk since it reduces intervention times of operators and thus the chances of controlling or mitigating the effects of an accident. A greater radionuclide inventory also has a direct impact in case of accidents since it will proportionately increase releases.

The Austrian Statement referred to a recent publication which showed that the plant operator is aware of the safety problems associated with power uprating, and also that open questions remain in this context [ELTER 2004].

In this publication, it was reported that deterministic and probabilistic safety analyses were under way in 2003/04. It was also stated that the impact of the proposed power uprate on core damage frequency (CDF) had not yet been quantitatively evaluated. It was assumed that heat removal success criteria would not be affected and it was believed that the time frame for successful operator response will not be significantly reduced. The impact on the frequency of large releases and on the progression of severe accidents had not been addressed.

The publication did not contest that safety margins are, in principle, reduced by power uprating. Also taking into account other sources, it was summarily stated that the most significant impact of the uprating results from the increased inventory and the possible acceleration of events in case of an accident.

A new type of fuel is mentioned in connection with the power uprating. It was pointed out in the Austrian Statement that no information concerning this fuel is provided in the PES. It was considered likely in the Austrian Statement that, following a marked trend world-wide, fuel with higher initial enrichment, and hence higher achievable burn-up, would be used.

3. Treatment of the Power Uprate Issue in the Environmental Impact Study and the Answers provided by Paks NPP

In the **EIS**, a new subsection has been added dealing with power uprate (section 2.2.5), which was not contained in the Preliminary Study. For better clarity and due to the importance of the topic, this chapter as well as the following one will be divided in subsections.

This division does not necessarily follow the structure of the EIS; subsections are defined here according to the relevance of various topics as seen by the authors.

Motivation for power uprating

Power uprating is planned in order to reduce production costs. Uprates by increasing thermal efficiency have already been performed (to 470 MWe). Further measures for efficiency increase would not be economical; therefore, a power uprate by increasing reactor power is now planned.

Margins relevant for safety

According to the EIS, results of analyses have clearly demonstrated that the power uprate will not lead to exceeding acceptance criteria. It is claimed that there is not even a decisive reduction of margins to the limits.

Relationship between power uprating and lifetime extension

It is necessary to coordinate the two projects of power uprating and lifetime extension. In this context, Paks NPP commissioned a feasibility study from VEIKI AG concerning the effects of a power uprate on the ageing processes of the main components of the units.

The result of this study was that power uprating would accelerate ageing processes; this, however, does not significantly influence lifetime extension. The effects of power uprating can be minimized by means which either are already implemented or will be implemented.

It is pointed out in section 3 of the EIS that the feasibility study was examined again after five years, in 2004/5, in the light of the additional operating experience gained in those years. In section 3.2.2, it is stated for a number of components like primary piping, safety valves, surge line etc. that deterioration processes will not be influenced by the power uprate. However, this statement is not explained or supported any further.

There is a more detailed discussion in the EIS concerning the reactor pressure vessels and the steam generators. Regarding the RPV, it is recognized that power uprating could, in principle, lead to an increase of the neutron flux in the RPV vessel wall. However, it is pointed out that due to the application of a Hafnium cover in the upper part of the control assemblies and to applying low leakage schemes, the neutron fluence at the internal vessel surface will actually decrease (see also Technical Item Reactor Pressure Vessel).

For the steam generators, it is pointed out that there will be exchanges of materials in the secondary circuit as a measure accompanying the uprate. Thus, erosion and corrosion in the secondary circuit should be reduced, and consequently, there should be less deposition in the steam generator tubes.

Safety assessments and plant modifications

It is explained in the EIS that the effects of the power uprating in case of design basis accidents had to be assessed. It had to be demonstrated that acceptance criteria will not be exceeded. This requires performance of a complete new set of safety analyses, taking into account changes in reactor physical parameters, actuation values of safety systems and in all other important parameters.

Those analyses also included calculation of the radioactive emissions and the resulting doses in case of design basis accidents, for the current power level (100 %) as well as for the increased power (108 %), to permit comparison.

The most important results are reported to be:

1. Thermo-hydraulic analyses demonstrate that in case of a large-break LOCA the integrated mass and energy flow is higher for 100 % power than for 108 % power – because of modifications of the hydro-accumulators.

2. Maximum pressure inside the containment is higher for 100 % power than for 108 % power.

The pressure of the hydro-accumulators has been reduced, and the amount of water they contain increased. After a power uprate, higher fuel element surface temperatures and higher oxidation of fuel elements are to be expected in case of LOCA. This effect is counteracted by making more coolant available in the hydro-accumulators. It is reported that the analyses show that this modification indeed achieves this purpose and leads to lower surface temperatures.

Further modifications which are mentioned are:

- Stabilizing primary pressure. The pressure control system of the pressurizer is to be modified, to keep primary pressure within a smaller bandwidth, at a higher level. Regulation will be dynamic instead of static.

- Modifications of main coolant pumps. The throughput rate of the main coolant pumps is smaller in Paks than in other VVER-440s, and different for different units. In the unit with the smaller throughput, it will be increased by fitting new impellers.

- Boron concentration in the primary circuit has to be increased.
- Improvements in reactor zone monitoring will be implemented to permit better control of the reactor.
- Turbines will be modified to allow for higher steam throughput.
- Modification of electrical systems. The cooling system of the generators will be modernized as well as some other backfits performed.

Fuel elements

According to the results of a feasibility study from 2001, the EIS reports, power could only be increased by three to four percent with the present type of fuel elements. Therefore, a new type of fuel has been developed.

The new fuel will be developed in two phases. In the first phase, a hafnium plate is welded to the upper part of the fuel element in order to achieve a more even distribution of subchannel temperatures. This is sufficient to achieve 108 % power. However, specific fuel utilization is reduced by four to five percent.

In the second phase, enrichment of the fuel will be increased. With this optimized fuel, better economics can be achieved, implementing a fuel cycle of five years. With the introduction of this fuel, the power uprate program will be concluded.

Radiological consequences of design basis accidents

The effects of the power uprate on the radioactive emissions of the NPP are treated in section 5.3.5 of the EIS.

Regarding normal operation, no changes are expected. Regarding design basis accident, it is stated in section 2.2.5 of the EIS that the inventory of the core as well as activity in the primary circuit are both increased when power is increased. However, it is claimed that this effect is offset by the modifications performed, in particular the increase of the water inventory of the hydro-accumulators.

Licensing procedure and implementation of the power uprate

The licensing procedure for the power uprate is a complex process with several steps. Every unit has to be licensed separately. The principal license was granted in November 2005. All modifications in connection to the power uprate are dealt with in a special licensing procedure. The modified fuel is already licensed and has been loaded into unit 4 in 2005.

There is already a schedule for the implementation of the power uprate, in two steps. They will be realized at different times in different units, permitting the experiences gained in one unit to be used for the next:

Unit 1: 103 % -2007; 108 % - 2008.

Unit 2: 103 % - 2008; 108 % - 2009.

Unit 3: 104 % - 2008; 108 % - 2009.

Unit 4: 104 % - 2006; 108 % - later in 2006.

In the **Answers** provided by Paks NPP, power uprating is also treated at some length.

Margins relevant for safety

It is said that as far as safety issues are concerned, it is a clear intention of Paks NPP to maintain the same level of safety as before. This is also required by the authority (HAEA) and is a pre-requisite of licensing the power uprating.

It is stated that the qualitative considerations show that the power increase will be performed without decreasing the safety margins of the units.

Relationship between power uprating and lifetime extension

At first it is stated that by law power uprating is a matter of HAEA licensing, based on modifications in the Final Safety Analysis Report, and has nothing to do with the Environmental Impact Study associated with lifetime extension. Moreover, power uprating will be licensed even if lifetime extension is not applied for. Since power uprating will precede the entire licensing process of lifetime extension, the lifetime extension process will be based on the plant parameters with increased power. Nevertheless, according to the Answers, it is probably not useless to provide information concerning the safety features of power uprating.

Furthermore it is stated that safety of the operation will not be deteriorated at all due to complex measures associated with the power uprating. Moreover, the complex measures of power uprating and those of lifetime extension represent a certain synergy, since the modifications in connection with power uprating will be implemented taking the needs of lifetime extension into consideration.

Since all loads, fracture mechanical and other parameters will be analyzed for the lifetime extension licensing process assuming 1485 MWth power of the primary circuit, there is no doubt that all the related special safety problems of the lifetime extension due to the power uprating will be considered during the licensing process.

Regarding the reactor pressure vessel, the statement of the EIS, section 3.2.2, is in effect repeated.

Safety assessments and plant modifications

The need to repeat the DBA analyses at increased power is recognized. These analyses have been completed and the results are already presented in the Final Safety Analysis Report. Thus the Accident Analysis chapter of the FSAR covers operation and accidents in the thermal power range from 1375 MWth to 1485 MWth.

A further plant modification will be performed together with power increase. The initial hydro-accumulator pressure will be decreased from 58.8 to 35 bar, with a simultaneous increase of their inventory by 10 m^3. This modification (which has been already performed at the Loviisa and Dukovany NPPs) is reported to have a clear positive safety effect: There will be better cooling in case of large-break LOCA scenarios. This modification alone has much larger safety consequences than the complex measures associated with the power increase.

Another important factor in the power uprating is the flow rate of the primary circuit. The higher the flow rate, the higher thermal power can be achieved. Nevertheless the increase of the flow rate may have negative effects, namely increased vibration and corrosion/erosion processes.

This is why it is not intended to increase the flow rates of the units above the design value, which is not reached at present. It will remain unchanged for units 1, 3 and 4. In case of unit 2, where the flow rate is significantly lower than the design value, it will be increased by replacing the impellers of the main circulating pumps.

It is also mentioned that there have to be some modifications in the secondary circuit.

Fuel elements

The main limitation of the reactor power lies in certain properties of fuel elements. The traditionally used fuel elements are licensed for use in a primary circuit with 1375 MWth power. The two limiting operation conditions, namely the maximum fuel pin linear heat rate and the maximum subchannel outlet temperature are fulfilled for any core configuration.

In the actual power increase project those two limiting conditions remain unchanged. Moreover, the nominal cycle length also remains unchanged (325 effective days). Under these circumstances, fuel economics will be worsened since more fresh fuel elements should be loaded into the reactor core than at present.

Furthermore, burn-up of the unloaded fuel assemblies will somewhat decrease. This leads to a certain decrease of the radioactivity in the primary water system, since this effect is reported to overcompensate the slight increase of the activity of corrosion products. It is also claimed that decay heat of the core will not increase proportionally to the power uprate.

Independent of the power uprating and its above consequences, another modification of the fuel assemblies is introduced, namely the application of a hafnium cover around the steel rod connecting the absorber and fuel part of control assemblies, which has again a positive effect on safety.

Radiological consequences of design basis accidents

The amount of radioactive materials potentially released during accidents will decrease, according to the Answers. Similarly, the amount of radionuclides and the decay heat will not increase proportionally with the power increase.

Licensing procedure and implementation of the power uprate

HAEA issued the modification licence-in-principle of the power increase in November 2005. Nevertheless power will be increased gradually, and the experience, gained at slightly increased power at the unit where it will be introduced first, will be evaluated and later on used in the subsequent steps of the project.

4. Discussion of Treatment of Power Uprate Issue in EIS and Answers

Motivation for power uprating

Currently a number of nuclear utilities are planning power uprates for their nuclear reactors and many of them have already gone through this modification process.

- Generally, **smaller power uprates** (up to 2 %) can be achieved by implementing enhanced techniques for calculating reactor power. This involves the use of more precise feedwater flow measurements, which, in turn, provide for a more accurate calculation of power.

- **Greater power uprates** (up to 7 %) usually involve changes to instrumentation set points, but still do not require major plant modifications.

- **Extended power uprates** that could go up to 20 % of the nominal power may require significant modifications, to major balance-of-plant (BOP) equipment.

The greater power uprates (less than 7 %) may require significant hardware changes such as refurbishment or replacement of equipment permitting a power uprate without violating any regulatory acceptance criteria. A detailed cost-benefit analysis needs to be performed, considering implications on various aspects such as safety analysis, both deterministic and probabilistic, handling of additional waste, spent fuel storage facility or reprocessing, environmental impact, etc. [IAEA 2004].

For NPP Paks it is aimed to reach the 500 MWe gross unit power (increase of 8%) only with utilization of modernized fuel (larger pin lattice) and with the implementation of certain relatively simple modifications (replacement of inlet wheels of high pressure turbine, modernization of the primary pressure control system and of core monitoring system) [BAJSZ 2002].

<u>Margins relevant for safety</u>

There are different types of margins which are used in the design and operation of nuclear power plants [IAEA 2004]:

- Safety margin: Distance between the safety limit and an acceptance criterion (regulatory requirement)

- Licensing margin (safety margin on the basis of analyses): Difference between an acceptance criterion and an analytical result for the corresponding parameter (determined in conservative manner)

- Analytical margin: Margin representing uncertainties in modelling and code.

- Operational margin: Distance between the operating envelope and the operational state.

- Design margin: Distance between the design criterion of a system or component, and the minimum value needed to reach the requirements.

The first four of those margins are closely related and can be regarded together as roughly describing the difference between operational state and safety limit (see fig. 1). The design margin is not related to any of them.

All of those margins are relevant for safety, in the sense that they all contribute to the distance to the safety limit, and they all provide certain reserves and robustness which can help to control a critical situation where unexpected problems occur or uncertainties turn out to be larger than estimated beforehand.

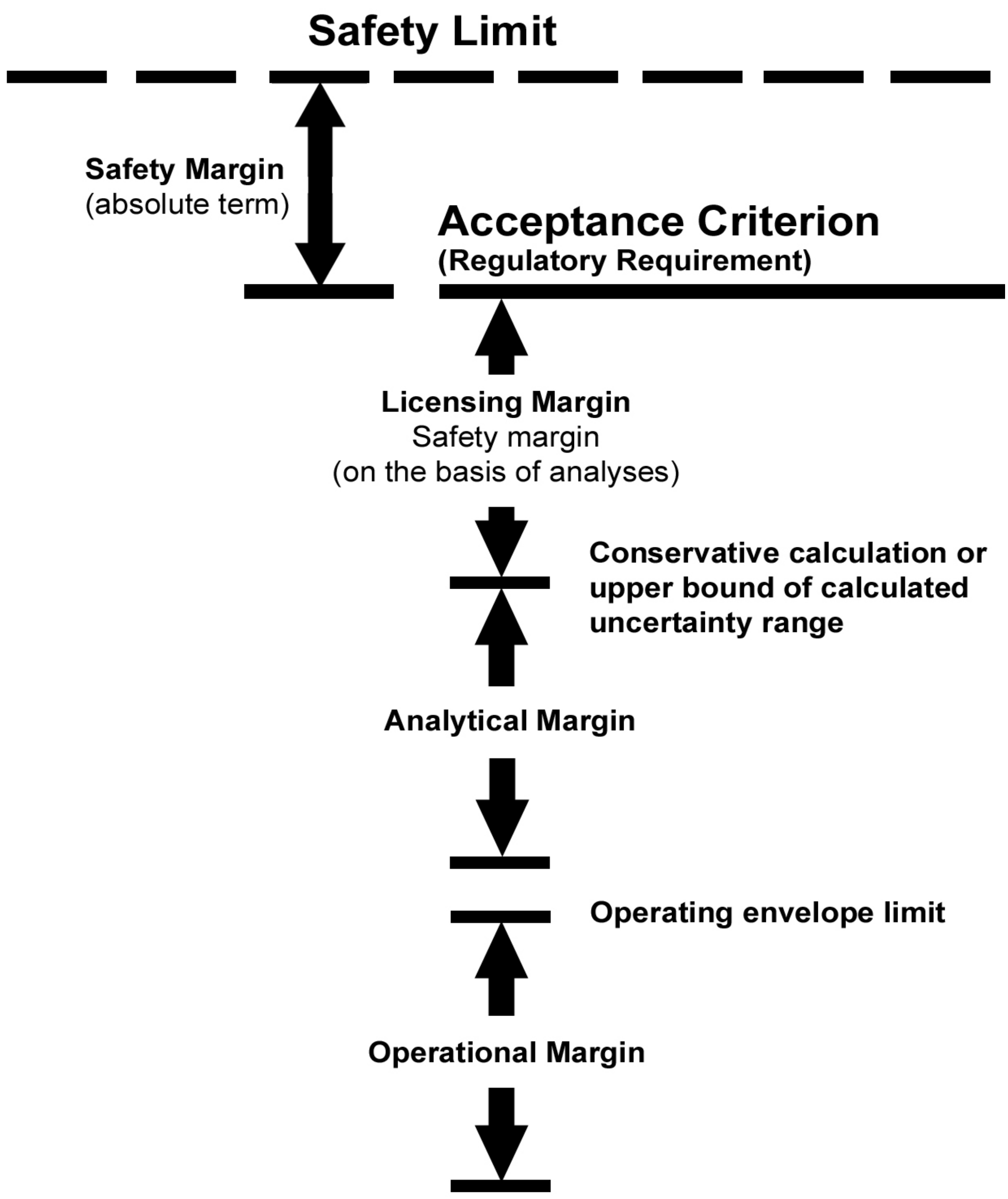
It is not always clear in the literature whether this terminology is strictly followed. In particular, it is not always clear when the term “safety margin” is used whether it refers only to the safety margin as defined above (first bullet), to the safety margin and the licensing margin which also constitutes a kind of safety margin, or also to some or all of the other margins which are, after all, in their entirety relevant for safety.

For Paks NPP, the EIS reports that there will be no “decisive reduction”, without specifying this claim. The Answers claim that there will be no reduction of safety margins.

Another source states that "[t]he planned power uprate should not assume significant reduction of any safety margin" [ELTER 2004].

These statements are somewhat contradictory, or they might indicate that different terminology is used in the sources quoted and that the Answers refer to the safety margin in the strict sense of the term, whereas the EIS and the other source refer to margins relevant for safety in general. This point requires clarification, both regarding the apparent contradiction and detailed information on which margins are reduced to which extent.

Fig. 1: Various types of margins for a nuclear power plant [IAEA 2004]



It is pointed out in the Answers that the two most important limiting operating conditions – the maximum fuel pin linear heat rate and the maximum subchannel outlet temperature – are still fulfilled after the uprate. This, however, does not exclude a reduction of the margin between the operating states and limiting operating conditions.

It should be kept in mind that any reduction of one of the margins mentioned will imply a decrease of safety. Margins which might appear overly conservative in the light of improved methods of analysis can still be useful to cover uncertainties. This is particularly important since plant operators generally can never be definitely sure that all uncertainties are covered. “Since the (modelling) uncertainties are not quantified and unresolved issues may also exist it is only a belief that the safety margin and the conservatism (which is an implicit margin) used in the operational parameters and system availability will cover the unquantified uncertainties” [ELTER 2004].

Relationship between power uprating and lifetime extension

The study performed by VEIKI AG mentioned in the EIS, concerning the effects of power uprating on the ageing processes, led to the result that power uprating would accelerate ageing processes – in agreement with the Austrian Statement. It is emphasized, however, that the lifetime extensions would not be significantly influenced by this. Not details are provided in the EIS; there is no discussion about what constitutes a significant influence.

For a number of components, the EIS claims that they will not be influenced by the power uprate, without any further explanation or specification. Reactor pressure vessels und steam generators are discussed to some extent. However, in those cases, too, the discussion remains on a general level. Measures are briefly explained without any quantitative specification or explanation.

In the EIS as well as in the Answers, it is stated that in spite of the higher power level in the reactor, the application of a low leakage fuel pattern actually permits a decrease of the neutron fluence in the pressure vessel wall (see also Technical Issue Reactor Pressure Vessel).

This is an interesting development since basically there is a certain conflict of goals between power uprate on the one hand (which implies an increase of the neutron flux in the reactor which should be distributed over the core as homogeneously as possible to avoid overly high peaking in individual fuel elements) and low leakage loading of the core (which implies low neutron fluence at the core periphery, and markedly higher fluence in the center of the core).

This point is not treated in the EIS and in the Answers. It is mentioned in a recent article on the new fuel type at Paks [KERESZTÚRI 2004], but with a very brief discussion only.

There is no discussion regarding synergetic or cumulative effects power uprating might have on components and systems.

In the Answers, it is explained that all loads, fracture mechanical and other strength parameters will be analyzed in the course of the lifetime extension licensing process assuming 1485 MWth power of the primary circuit. This indicates that the statements in the EIS have to be taken as preliminary and that definite and meaningful results will only be available after the analyses in the context of the licensing procedure have been performed.

<u>Safety assessments and plant modifications</u>

The EIS mentions some modifications which have been performed or are about to be performed. They concern the hydro-accumulators, the pressure control system, the main coolant pumps, the boron concentration in the primary circuit, reactor zone monitoring, turbines and electrical systems. In addition, the Answers mention some unspecified modifications in the secondary circuit.

A power uprate creates a new thermal balance of the plant. A comprehensive investigation of all systems and components concerned has to be performed in this context, for example regarding:

- Reactor core (design and control)
- Primary circuit with pressurizer relief system
- Secondary circuit with secondary overpressure protection systems
- Emergency core cooling system, emergency feedwater system
- Information and control systems
- Electrical systems

Some of these systems and components are addressed in the EIS and the Answers. However, there is no comprehensive discussion of all systems and components which could be concerned by a power uprate.

For example, modifications of the hydro-accumulators which are important in case of a large-break LOCA are mentioned. However, there is no discussion of modifications which might be required in other parts of the emergency core cooling system, like the high-pressure injection which is important for controlling small-break LOCAs, or the containment sprinkler system which reduces containment pressure in case of LOCA.

For power uprating, it has to be shown that design basis accidents can still be controlled. Accordingly, it is stated in the EIS and the Answers that a complete set of analyses for design basis accidents has already been performed, both for 100 % and 108 % power. The DBAs which were analyzed are presented in the EIS.

However, clarifications are still required in this context. In table 5.5.12 of the EIS, the DBAs which were analyzed for resulting emissions are listed (12 in all). In eight cases, it is noted that the analysis was performed for 108 % power; in four cases (Nos. 6, 10, 11 and 12), the power level is not specified. It should be clarified whether this is due to an incomplete translation, a simple erroneous omission in the EIS, or the fact that these DBAs indeed have not been analyzed for 108 % power.

<u>Fuel elements</u>

In the EIS, the two-phase process of the development of new fuel elements is described; it is explained that longer cycle times (5 years) and higher enrichment will be realized in the second phase. This will necessarily lead to higher burn-up.

The Answers are incomplete in this respect. They only mention the first phase with reduced burn-up and hence, a reduction of decay heat and of radioactivity in the primary circuit. This omission is somewhat misleading since it could give rise to the impression that no further phases are planned and burn-up will not be increased later.

The longer-term tendency towards burn-up increase has already been assumed as likely in the Austrian Statement.

No schedule is given for the second phase of the introduction of a new type of fuel. Since the power uprate will only be completed with this second phase, it can be expected to be implemented rather soon.

Radiological consequences of design basis accidents

In case of power uprating the core inventory of radioactive nuclides changes: The inventory of short-lived fission products is proportional to the reactor power whereas the inventory of nuclides with longer half-life is proportional to the fuel burn-up of the core. Thus, after the first step of introduction of a new type of fuel, core inventory of short-lived fission products will increase, whereas the inventory of long-lived radionuclides will somewhat decrease. After the second step, the inventories of both categories of radionuclides will be increased, compared to the present situation.

It has to be emphasized that in the time frame of interest for accidents (up to several days), the decay heat is essentially determined by the decay of the short-lived radionuclides. Hence, for this important period of time, decay heat will in fact increase proportionally to reactor power, in contrast to what is claimed in the Answers, already in the first phase of new fuel introduction.

Regarding radioactive releases in case of an accident (DBA or BDBA), the long-lived radionuclides are of dominant importance. In this respect, in the first phase of new fuel introduction, the source term for accidents will indeed be reduced. However, this will be reversed in the second phase when higher burn-up will lead to a higher core inventory of long-lived radionuclides, too.

Licensing procedure and implementation of the power uprate

It is appropriate to implement a power uprate in two steps, to be able to apply the experience gained with the first step in the planning and implementation of the second. However, in the case of Paks NPP the steps are closely following each other – they are to be implemented in subsequent years for each unit, and within 4 years for all units.

It remains doubtful whether it will be possible to collect and apply operating experience within this comparatively rapid sequence of events.

It is interesting to note that at German PWRs of the 3rd and 4th generation, smaller power uprates (below 5 %) were implemented considerably slower in two steps, with an increase by about 2 % at first, followed by an increase by about 2.5 % after 7 to 10 years.

Power uprating and beyond design basis accidents (BDBAs)

The consequences of power uprating for the course of BDBAs also have to be considered. This topic is not treated in the EIS and the Answers; all references to accidents there concern design basis accidents.

However, what has been said concerning the radiological consequences of DBAs is also relevant in the context of BDBAs, particularly regarding the increase of decay heat in the core and the increase of the core inventory of long-lived radionuclides in the second phase of introduction of new fuel elements.

The increase of decay heat leads to reduction of the intervention times available for severe accident management measures. For German PWRs, for example, it has been shown that this reduction can be significant [RSK 2003]:

For example, in case of complete loss of feedwater, immediate availability of personnel and immediate shutdown of the main coolant pumps, the time available for the initiation of secondary side pressure relief is reduced from 13 to 9 minutes, and for primary side pressure relief from 27 to 21 minutes.

In case of station blackout, the corresponding reductions are from 29 to 21 minutes, and from 26 to 19 minutes.

In realistic circumstance (for example if personnel is not available right away), those time spans are still reduced further. All in all, the probability for the double failure of secondary and primary pressure relief will be increased by a factor of 2 to 4.

In a Hungarian source, it is stated that the time frame available for successful operator response will not significantly be reduced. It was further reported, however, that the most significant impact of the power uprate results from the increased radioactive inventory and the possible time acceleration of events due to the increased decay heat level and corresponding decreased time to core uncovery [ELTER 2004].

Results of projects for safety reassessment of the Paks NPP show that among the factors determining the core damage, the effect of human errors is the most significant [SZABADOS 2002]. The reduction of intervention times is likely to increase the probability of such errors in the phase of severe accident management.

There are clear indications that analyses of BDBA scenarios have been performed or are planned to be performed for Paks NPP. It appears that the source quoted above [ELTER 2004] is referring to preliminary considerations in this respect. The same source reports that "[t]he impact on large fission product release frequencies and on progression of severe accident(s) has not been addressed yet". However, no information has been provided on this subject in the EIS and the Answers.

Regarding releases, it is to be expected that they grow more than proportionate to the power uprate. For example, for the German PWR Grafenrheinfeld, it has been reported that a power uprate by 4.9 % would lead to an increase of the inventory of 6.2 % [SSK 2003].

There are similar results for the Swiss NPP Leibstadt (BWR). All in all, a 14,7 % increase in power was estimated by the regulatory authority to lead to an about 30 % increase in the risk of release of activity. This high increase in releases is due to the fact that not only the higher inventory leads to higher source term in this case; also the time acceleration of events because of higher decay heat, and hence the earlier occurrence of containment failure [DOESBURG 2004]

5. Information Provided by the Hungarian Representatives at the Hearing June 6, 2006

At the **Hearing,** the Hungarian side provided no further information regarding power uprate.

The Hungarian experts explained that the Safety Report contains all analyzed DBA and BDBA sequences (including PSA level 2). However, it did not become clear which accident sequences were performed on the increased power level.

6. Assessment

More information concerning safety margins and other relevant margins, in the context of the power uprate, would be of interest to the Austrian side. Clarification of apparently contradictory statements about the reduction of margins would be desirable, as well as detailed information on which margins are reduced to which extent.

The combined effects of introducing a new type of fuel and the power uprate on the neutron fluence in the reactor pressure vessel wall, including a discussion of the potentially conflicting goals involved, would also be of interest.

Furthermore, a comprehensive discussion of all systems and components which could be concerned by a power uprate would also be of interest.

The uncertainty remaining concerning the DBA analyses performed should be resolved.

The two phases of fuel development also are of interest to the Austrian side and should be explained in more detail; particularly concerning the schedule of the second phase, the burn-up which will be achieved then and its possible effect on source terms for DBAs and BDBAs.

The claim expressed in the Answers that the decay heat of the core will not increase proportionally to the power uprate should be further explained and supported both for the first and second phases of fuel development.

The planned schedule for implementation of the power uprate should be discussed in more detail, particularly regarding the question to which extent it permits collection and feedback of operating experience.

Of particularly great interest to the Austrian side are all questions concerning the potential effects of the power uprate in case of BDBAs – mainly, reduction of intervention times and changes in the source term.

At the **Hearing** no further information concerning the critical aspects of power uprate brought up by the Austrian experts was provided.

It is not clear that all relevant DBA and BDBA sequences were performed on the increased power level. All considerations should be based on the uprated power level (108 %) since the power uprate will be completely implemented at unit this year at all units within a few years.

As an important first step, provision the parts of the Safety Report relevant for understanding the assessment of accident sequences referred to at **the Hearing** by the Hungarian side would be helpful for the clarification of the open questions.

TI Spent Fuel Storage

1. Introduction

Spent fuel arising at Paks NPP is being stored in a Modular Vault Dry Storage facility (MVDS facility) at the site, after it has been removed from the reactor pools.

In this storage facility, fuel elements are enclosed in fuel storage tubes, which then are inserted into a silo built of massive concrete, where they are stored in vertical position. Heat is removed by passive cooling (thermal circulation of air). There is no high pressure in the tubes. Hence, there are hardly any driving forces for accidents with large releases, due to internal events.

On the other hand, the inventory of spent fuel in the storage facility is considerably larger than the inventory in the reactor cores and the reactors' spent fuel pools at the Paks site today, and will grow still more in the coming years, particularly if plant lifetime is extended. This implies that there are considerably larger quantities of long-lived radionuclides in the spent fuel store. Therefore, it seems appropriate to briefly treat the issue of potential accidents due to external influences on the storage facility.

The question of possible consequences of contamination of the store due to a reactor accident will also be considered, as well as terror attacks. Finally, the issue whether a final repository will be available in time will be briefly discussed.

Fuel storage was not discussed in the Austrian Comments of September 2005, and therefore not treated in the Answers provided by Paks NPP. It has been mentioned in the Preliminary Environmental Study and also in the Environmental Impact Assessment, where some information on the present stage of construction and the planned, now imminent enlargement was provided.

2. Treatment of the Storage Issue in the Preliminary Environmental Study and the Environmental Impact Assessment

It was described in the Preliminary Environmental Study and the EIS (section 2.2.2.3) that 400 – 460 spent fuel elements are produced per year in the four Paks units (one fuel element contains 0.117 t of heavy metal). Minimum cooling time in the reactors' spent fuel pools is three years.

Until 1998, 2.331 fuel elements have been transported to the Soviet Union, later to Russia for reprocessing. Hungary is not required to take back the radioactive wastes from reprocessing.

In 1991 it was decided that it might not always be possible in the future to transport spent fuel to the Soviet Union/Russia. It was decided to prepare a Hungarian solution for spent fuel management. The MVDS facility was constructed by GEC Alsthom and received the first fuel in 1997.

The EIS reports that at present, there are 11 storage modules for 450 fuel elements each. In mid-2005, enlargement was begun (construction of five more moduls).

Storage is envisaged for a period of 50 years, which, it is reported, should provide sufficient time for far-reaching decision concerning final disposal.

3. Discussion of Treatment of Storage Issue in the Environmental Impact Study and Answers[2]

The 11 storage modules being available at present will contain a total of 4950 fuel elements. At the end of the lifetime of Paks NPP as presently envisaged (30 years of operation), there will be need for a storage capacity of at least 11,000 fuel elements (this number can be influenced by the fuel strategy applied) [ÖRDÖGH 2004].

According the EIS, section 2.2.2.1, there are 312 fuel elements in every reactor core, giving a total of 1248 fuel elements in all cores. In addition, there will be about 1290 spent fuel elements in the four reactor pools, if fuel is stored for the minimum period of three years. Thus, the inventory of the MVDS store will be up to nine times the inventory of the reactor cores, and up to four times the inventory of reactor cores plus reactor pools, if the fuel inventory in the pools corresponds to the minimum required cooling period.

Compared to cask storage of spent fuel, as practiced in many countries (for example, in Germany, the Czech Republic and the USA), the MVDS concept is more prone to disturbances and incidents. Considerably more fuel storage tubes than casks are required for a given number of fuel elements. Hence, loss of tightness due to corrosion, material flaws etc. is likelier. Loading and unloading of the tubes into the silos, which is performed with a handling machine positioned above the silos, is more risky than handling of storage casks; in case of the fall of a filled storage tube, the fuel elements will be more directly concerned.

Regarding external impacts, the MVDS concept also appears to be less robust. Seismic design of silos and tubes is more complicated than the design of casks; this also applies to pressure waves from external explosions.

In case of an accident in one of the reactor units, leading to radioactive releases, the silos and the outer surfaces of the storage tubes could be contaminated through the natural draft air cooling of the store. Decontamination does not appear possible; the interior of the silos is not accessible and even if it were, the high dose rates due to direct radiation would hardly permit decontamination activities.

In case of further operation of the storage facility after a contamination event, there would be continuous radioactive releases, because the contamination attached to the storage tubes would be partially removed and transported to the atmosphere through the circulating air. Thus, the fuel in the store would have to be evacuated after it has been contaminated which would require a complicated and lengthy procedure.

There is no information in the Preliminary Study and the EIS concerning the protection of the fuel store against external impacts. It seems likely that the storage facility is not

2 The content of this section, regarding incidents, external impacts, possible influences of a reactor accident and terror attacks, is based on a contribution by Wolfgang Neumann, Gruppe Ökologie Hannover.

designed against the crash of a commercial aircraft, or various conceivable terror attack scenarios like shelling.

In any case, the upper part of the storage buildings, with the fuel handling machine, would be destroyed in case of an attack. Thus, access to the fuel elements would be denied because the silos would be covered by debris.

Generally, the likelihood of a destruction of all barriers which could prevent releases appears to be greater for the MVDS concept, than for cask storage. There is a larger vulnerable area for the hard, penetrating components of an aircraft (for example, engines and landing gear) to hit, leading to destruction of the main barrier (walls and roofs of silos). If this barrier fails, many storage tubes will fail in consequence. Up to 450 fuel elements can be affected (in case of a storage facility with CASTOR VVER casks, 84 fuel elements can be affected). Therefore, it is to be expected that releases will be larger than for the cask concept.

This will be particularly valid if there is a kerosene fire, as must be expected, since the remnants of a silo will constitute an “ideal” tub, where kerosene will collect and burn on a relatively small area.

Counter-measures after the crash of an airliner, as well as measures for evacuating the store, will be more complicated, since the sources of releases are more spatially distributed, and the structure of the partially destroyed buildings are more complex.

Because of the larger vulnerable area, attacks with armour-piercing weapons, missiles or artillery pieces will require less accuracy. On the other hand, it is likely that the number of fuel elements concerned will be smaller than in case of penetration of a cask, if there is no explosion inside a silo.

The first fuel elements were loaded into the storage silos in 1997. As has been reported, a storage duration of 50 years is envisaged, which is regarded as providing sufficient time for decisions on final disposal.

According to the National Report of Hungary in the framework of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [HUNGARY 2005], a final repository is due to be operational by the end of the 2040s. If the storage duration of 50 years is not to be exceeded, this planning leads to a very tight situation since the first fuel would have to be unloaded from the storage facility in 2047. Furthermore, the realization of a final repository for high-activity wastes is notoriously complicated and it cannot be excluded that there will be delays, possibly even for decades. Thus, there is no guarantee that the planned storage duration of 50 years can be kept; it could be exceeded for a considerable part of the store’s inventory.

4. Assessment

The fuel storage facility at the Paks site already contains a large amount of radioactive materials, which will grow considerably in the coming decades if lifetime extension is implemented.

The storage concept employed appears to be more vulnerable to external impacts and terror attacks, than the cask storage concept employed in many countries. Furthermore, it is likely that it will pose more problems in case of contamination of the store through a reactor accident.

Therefore, further information regarding the storage concept's vulnerabilities, and the possibilities of large releases from the store, would be of interest from the Austrian point of view.

There are indications that the planned storage duration (50 years) is likely to be exceeded. A discussion of this point, in the context of the planning for a final repository in Hungary, would also be of interest from the Austrian point of view.

Technical Issues: Summary of Further Information of Interest from the Austrian Viewpoint

In the sections dealing with the eight Technical Issues, further information which would be of interest for assessment of severe accident risk with transboundary emissions has been identified and compiled in the last subsection.

Below, the information identified is listed in a summarized manner.

<u>TI Ageing Management Program</u>

- As a first helpful step, provision of the information reported orally at the Hearing in written form.
- Further information concerning the changes and developments of the ageing management program during the coming years.
- In particular, further information on the new approach to in-service-inspections to be introduced at Paks, which is to include reductions in inspection efforts without a decrease of the safety level.

<u>TI Reactor Pressure Vessel</u>

As a first step, provision of an English version of the Hungarian guideline regulating the RPV safety assessment (No. 317) would be helpful; as well as more detailed specification in which context and to which extent other regulations (e.g., VERLIFE and ASME) are being applied. However, in addition to the regulatory framework, concrete information regarding the implementation of RPV safety assessment at Paks would be of great interest.

- Information concerning the database for un-irradiated materials, a comprehensive description of the surveillance program, information on the scope of the PTS analyses, the thermo-hydraulic calculations already performed, and the methodology to be applied regarding fracture mechanics.
- Information concerning the safety margin to be applied in the PTS analyses, as well as assumption on cracks, on RPV cladding integrity, and on the manner of WPS effect application.
- The possible effects of a dose rate effect and the way this effect is investigated in the surveillance program.

- Proof for the claimed reduction of neutron fluence in the vessel wall in spite of power uprating. (see also TI Power Uprating)
- Possible consequences of changes in the emergency core cooling system for the course of accidents.

As the new PTS analyses will be performed in the coming years, further information on methodology and results would be of great interest to the Austrian side.

TI Steam Generators

As a first step, provision of the information reported orally at the Hearing in written form would be helpful. Depending on the content (some of which might not have been transmitted optimally at the Hearing, due to the limited time available and the limitations of simultaneous translation), further information might be of interest regarding:

- The role of the Nickel content in stress corrosion cracking of steam generator tubes.
- Erosion-corrosion of the feedwater system and the measures taken in this context.
- Information on the follow-up work on the accident in 2003, particularly regarding aspects relevant to corrosion problems.

In-service-inspections will be developed further in the next years. Information permitting to follow the new developments and modifications will be of interest from the Austrian point of view.

TI Confinement System

- More detailed information on ageing of the barbotage condenser system as well as on backfitting measures which were performed in the last years or are planned for the immediate future (particularly in connection to the PHARE experiments and investigations), and the reduction of the leak rates.
- Time-pressure curves in the confinement for DBAs, including variations of the two cases analyzed for maximum ECCS configuration.
- Information to which extent tests and investigations were performed or are planned regarding the capabilities of the confinement system in case of beyond design basis events.
- Discussion of the behavior of the confinement system in case of a BDBA, including a discussion of safety reserves and of capabilities for accident mitigation, time-pressure curves and a discussion of the consequences of the leak rate for the timing and extent of releases.
- Discussion of planned and possible backfitting measures to improve the mitigating capabilities of the confinement system in case of BDBA.

As the issue of hydrogen recombiners for BDBAs is further discussed and developed, it would be of interest for the Austrian side to follow investigations and, if a decision for installation is taken, planning and implementation of measures.

TI Seismic Hazards

- Explanation and detailed discussion of the high value which is at present given for the core damage frequency due to seismic events.

- Information concerning the current state of seismic backfitting.

- Discussion and an estimate of the reduction of CDF achieved so far, and to be achieved by further backfitting.

New investigations of seismic issues, including a new assessment of seismic hazards, will be performed in the coming years. Due to the considerable contribution of seismic events to the overall risk at Paks, it will be of interest from the Austrian point of view to closely follow those investigations and assessments.

Seismic backfitting activities, too, are of considerable interest and should be followed in the coming years.

<u>TI Terror Attacks</u>

Considering the potentially grave consequences, the issue of terror attacks is of great interest to the Austrian side, especially due to the current increase of the general terrorist threat in Europe, acknowledged by the Answers.

Vulnerabilities, attack scenarios and potential consequences can and should be discussed in an appropriate general manner, and in an appropriate setting. Regarding public debates, the criterion applied should be that it would be pointless to attempt to keep such information secret which a competent group of attackers can easily acquire.

<u>TI Power Uprating</u>

- More information concerning safety margins and other relevant margins, in the context of the power uprate.

- Clarification of apparently contradictory statements about the reduction of margins, as well as detailed information on which margins are reduced to which extent.

- The combined effects of introducing a new type of fuel and the power uprate on the neutron fluence in the reactor pressure vessel wall, including a discussion of the potentially conflicting goals involved.

- Comprehensive discussion of all systems and components which could be concerned by a power uprate.

- Information to clarify the remaining uncertainty which different power levels have been taken into account in the DBA and BDBA analyses presented in the EIS and at the Hearing.

- Detailed information on the two phases of fuel development; particularly concerning the schedule of the second phase, the burn-up which will be achieved then and its possible effect on source terms for DBAs and BDBAs.

- Substantiation of the claim expressed in the Answers that the decay heat of the core will not increase proportionally to the power uprate for the first and second phases of fuel development.

- The planned schedule for implementation of the power uprate, particularly regarding the question to which extent it permits collection and feedback of operating experience.

- Information on all questions concerning the potential effects of the power uprate in case of BDBAs – mainly, reduction of intervention times and changes in the source term.

TI Spent Fuel Storage

- Further information regarding the storage concept's vulnerabilities, and the possibilities of large releases from the store.
- Discussion of the planned storage duration (50 years), in the context of the planning for a final repository in Hungary.

PART II Accidents and Emissions

Introduction

In the **Statement** on the Preliminary Impact Assessment Study of September 2005, a legal statement explains why information about severe accidents, i.e. Beyond Design Base Accidents and discussion of mitigation measures has to be part of the Environmental Impact Assessment. To exclude severe accidents in the EIA documents is neither in line with the EIA directive nor with the ESPOO-Convention.

In the **Ruling** of May 2005, the competent Hungarian environmental authority demands that the detailed environmental study has to provide the analysis of events which could cause transboundary emissions.

Chapter 5 of the **Environmental Impact Study (EIS)** covers initiating events, Design Base Accidents (DBA) and the probability of severe accidents. Chapter 8 summarizes the method and results of dose assessment caused by releases from design base accidents. Chapter 10 disputes the requirement to provide information on transboundary impacts in the Environmental Impact Study.

Information and Statements of these parts of the EIS, the corresponding **Answers** of Paks NPP to the Statement and information provided at the public **Hearing** are discussed in the following sections

Core Damage Frequency & Severe Accidents

According to the **Preliminary Environmental Study (**PES**),** the overall risk for core damage in Paks is estimated to be between 3.4 10^{-5}/year for unit 2 and 4.5 10^{-5}/year for unit 3 (PES table 5.50) - but this core damage frequency evaluation considers only internal initiating events. A CDF of some 10^{-5}/year is comparable to the older NPPs in Germany; later built German NPPs reach a CDF of some 10^{-6}/year, i.e. the accident risk of these plants is lower by a factor of 10 than that of Paks NPP.

According to the **Environmental Impact Study** the total CDF is 3.0 10^{-4}/year. The dominating contribution, with 86% of the total, is the seismic risk (Figure 5.5.1). The **Ruling** states that severe accidents can be excluded from the evaluation if the probability is less than 10^{-5}/year. All CDF evaluations presented in the documents exceed 10^{-5}/year. Moreover, taking into account the seismic hazard, the resulting CDF exceeds the limit of 10^{-5}/year substantially.

The value of 3.0 10^{-4}/year is also considerably higher than the target for severe core damage frequency for existing nuclear power plants set by the IAEA's International Nuclear Safety Advisory Group (1.0 10^{-4}/year) [INSAG 1999].

The CDF of 2.58 10^{-4} per unit and year for seismic events alone constitutes a very high value. An earthquake will hit all 4 units at once and therefore can lead to severe accidents in all four reactors simultaneously. Moreover all other facilities at the site, including the spent fuel store, would be endangered by an earthquake as well. After an earthquake the situation at the plant could be extremely complex and confusing.

Obviously there is the potential for a large release of radioactive substances into the atmosphere and water.

The **Environmental Impact study** reports that improvements of structures and protection of safety relevant equipment against earthquake hazards is ongoing. Concerning the measures and the reductions of CDF which might already have been achieved, no details are available. Detailed information and discussion of the earthquake risk and the related improvements of seismic resistance of the plant is of high interest for Austria.

At the public Hearing in Mattersburg the Hungarian side presented their latest results of safety analysis:

Due to the first reconstruction measures in Paks NPP a reduction of seismic risk to a CDF of 6.6 10-5/year was achieved. Although this is a substantial improvement, the units of Paks NPP are only just meeting the IAEO target value of 1.0 10-4/year.

Regarding BDBAs the Hungarian side explained that 80% of the analyzed BDBA – sequences lead to a cesium-release of less than 1% of the inventory and only 6% to a release of more than 20%.

There is no justification to exclude severe accidents from evaluation in the environmental impact assessment in principle according to the ESPOO Convention and the EIA directive. Based on the data presented in the EIS, the evaluation of severe accidents and the related cross-border emissions is also required by the Hungarian regulations as they are described in the Hungarian documents (CDF > 10^{-5}/year).

The **Answers** to the Austrian Statement refer to the Level 2 PSA study which was carried out between 2000 and 2004, where it was pointed out that the Paks units have larger reserves (water, concrete, steel) against core melt than other reactor types. The containment study referred to in the **Answers** is reported to show that the real capacity of the containment is much higher than design value. But these investigations concerned solely design base accidents. Regarding protection against severe accidents, it is noteworthy that the WENRA evaluation concludes "that the containment capability of VVER plants to limit releases appears to be somewhat inferior to Western PWR containments“ [WENRA 2000].

The **Answers** explain further that the Level 2 PSA led to acceptable results partly due to the preventive accident management measures introduced by the plant several years ago: "The strategy of the accident management is to prevent core melt even in the case of beyond design base accidents." More actions are planned and some of these shall be implemented during the licensing process of lifetime extension. A second group of measures, including technical modifications, shall be implemented if lifetime extension is approved.

Additional information about the Level 2 PSA and the planned measures of emergency preparedness and technical modifications are not available in the documents.

Design Base Accidents

Chapter 5 of EIS describes the accident analysis method which follows in principle the US NRC regulatory Guide 1.70.

In **chapter 5 and 8 of EIS,** loss of coolant accidents (LOCA) in the primary system within the confinement und scenarios with containment bypass scenarios (PRISE) are described and their impact analyzed. The scenario description is not very detailed and the alleged conservatism of assumptions is not fully comprehensible.

Pipe break analyses for the hot and cold side of the primary cooling system (including the double ended guillotine break of a large pipe) were made under the assumption that a minimum configuration of emergency cooling system is available (failure of 1 of the 3 ECCS systems). The **LOCA** consequence analyses were made under the assumption that the containment spray and the bubble condenser are working. It was shown that a LOCA Ø 492 (hot side) does not lead to emissions beyond the limits if 2 of the 3 spray systems are working.

Containment bypass accident scenario: failure of steam generator collector leads to opening of the safety valves. If these valves do not close because of two phase and water flow, primary coolant is released into the atmosphere and if it is not possible to stop the release the core falls dry. To stop the process operator intervention is required (EIS, 5.5.2.3.1).

Steam generator collector failure at first was regarded as beyond design base accident. Due to the VJB classification system it was required to be evaluated as a DBA because of the achieved safety improvement (PES chapter 8). Operator intervention, in principle, is an appropriate reaction but it cannot guarantee that such an intervention is possible under all accident conditions.

In chapter 8 of the EIS, a reconstruction is presented in order to redirect the effluent of coolant from the primary side into the confinement. If it is possible to realize this reconstruction as planned in 2008, this would an unusual solution to mitigate the accident risk.

Additional information about the accident sequences and time tables after power uprate are required, as well as about the assumed status of essential systems in order to understand that low probability of large release of radioactive substances can be achieved.

Chapter 8 of the EIS also discusses a program for mitigating the impact of severe accidents (BDBA). This program contains accident management and emergency guidelines as well as reconstruction measures, including the installation of an independent power source for the safety valve of the pressurizer and the installation of hydrogen burners. This program has been accepted and shall be implemented in accordance with the lifetime extension.

Source Terms:

EIS presents source term data and dose assessment for DBAs. The much larger release in case of a BDBA (severe accident) is not discussed at all.

It is difficult to compare the source terms presented in the Preliminary Environmental Study to the ones presented in EIS because of the very brief descriptions of the accident sequences.

The transboundary accident risk is regarded as negligible in the EIS. This is justified by the dose assessment of design base accidents, carried out by PC COSYMA because the DBA source terms do not lead to significant exposure outside the radius of 24 km.

However, the magnitude of the release and the distribution of emissions in case of severe accident are not comparable to the controlled release of substances through the filter system in case of a design base accident.

Questions of the confinement capability to prevent releases in case of a core damage are of high relevance for the discussion of DBA and their potential to turn into a severe accident. In this connection the following accident sequences should be discussed (Tab 5.5.9 - 5.5.12):

- Breaks of 492 diameter pipes cold and hot site at normal operation and at low power (during shut down).
- Containment bypass scenarios at 108% power.

The point of this discussion is the maximum pressure in the confinement due to the LOCA compared to the maximum design pressure of confinement, the reaction of the barbotage tower, particularly during the DBA scenarios were maximum ZÜHR configuration is supposed. The description of the DBA sequences in chapter 5 is confusing because of the mix of numbers and titles, different conditions of ECCS and confinement configurations.

At the **Hearing** in Mattersburg the Hungarian side explained that the Safety Report contains all analyzed DBA and BDBA sequences and the corresponding results - instead of the EIS, which refers only a part of the results. In order to evaluate the risk of severe accidents and to understand the planned improvement measures it would be useful to get access to the parts of the Safety Report relevant for accident analysis.

At the **Hearing** a Hungarian representative argued that transport of released radionuclides after an accident to Austria would be very unlikely due to the weather conditions (3-5%) were disproved by a study financed by the Austrian Federal Ministry of Agriculture, Forestry, Environment and Water Management. This study shows that the probability for transport of radionuclides released by a potential accident at Paks to Austria, and deposition exceeding Austrian intervention levels is 28% for level 2 and 15,7% for level 3, respectively - requiring protection measures for children (level 2) or for the children and adults (level 3).(SEIBERT 2004) Differences in the assessment of long-range transport of emissions are explained by different models used by Austria and Hungary. PC COSYMA is a tool for assessing radiological impact only in the vicinity of the plant).

Conclusion:

It is acknowledged in the EIS that severe accidents (BDBAs) cannot be excluded; this follows implicitly from the results on CDF as presented. In this statement, various questions have been raised concerning potential problems with safety relevance, which further emphasize that BDBAs are possible. Effects of power uprate as higher core inventory and acceleration of accident sequences, as well as ageing effects, reduce the safety margins of the plant. Thus, it is clear that the discussion of potential transboundary effects cannot be restricted to DBAs.

The release of radioactive substances can affect regions in a distance of several 100 km from the source. The Austrian Statement of September 2005 presented the assessment of potential impacts of a BDBA for different weather situations in Central Europe. In this assessment, the release of 30% of the Cs-137 inventory of one reactor core was assumed. In case of an earthquake or terror attack, the release could be even higher.

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Glossary

AMP	Accident Management Program
BDBA	Beyond Design Base Accident
Bq	Becquerel, unit for radioactivity
CDF	Core Damage Frequency
Cs-137	Cesium 137
DBA	Design Base Accident
ECCS	Emergency Core Cooling System
ECT	Eddy Current Testing
EIA	Environmental Impact Assessment
EIS	Environmental Impact Study
GWd	Gigawattday – unit for energy
I&C	Instrumentation and Control
IAEA	International Atomic Energy Organization
ISI	In-Service Inspection
MVDS	Module Vault Dry Storage
NPP	Nuclear Power Plant
PES	Preliminary Environmental Study
PSA	Probabilistic Safety Analysis
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RPV	Reactor Pressure Vessel
SG	Steam Generator
SSC	Systems, Structures and Components
Sv	Sievert, unit for dose equivalent
TI	Technical Issue
VVER	Russian Version of the Pressurized Water Reactor
WPS	Warm Pre-Stressing

ANNEX:

Treatment of requirements stated by the Ruling

Besides a serious assessment of severe (accidents BDBAs), most of the Ruling's requirements have been met by the Environmental Impact Study, even if not all the explanations are detailed enough to allow a meaningful evaluation and some of the requirements could not be fulfilled because the assessment is still ongoing (seismic).

I. General Legislation:

A pt. 27 demands that the detailed environmental study has to provide the analysis of transboundary effects and measures for a minimization of these effects. **A pt. 27 b)** requires the description of events which could cause transboundary impacts. **A pt. 27 c)** requires an assessment of the impact of transboundary emissions **A pt. 27 g)** methods and data concerning the impact assessment

II. Radiation Protection:

pt 1 b) requires to assess the storage capacities for radioactive waste and to discuss alternative solutions for radioactive waste management **pt 3 b)** demands the dispersion analysis of emissions of the design base accident with the maximal radioactive emissions and the impact of this release on environment and health of the population **pt 5** demands the description of the power uprating and the potential increase of radioactive emissions. **pt 9** requires the analysis of the incident in PAKS unit-2 on April 10, 2003. Regarding this latter point, it has to be pointed out that the information in the EIS is incomplete. The underlying issue of steam generator corrosion and subsequent fouling of fuel elements is not discussed in the context of the incident.

XI Specific Regulations:
pt 4 requires a detailed geological assessment and the evaluation of the seismic design of the plant.