ENVIRONMENT AGENCY AUSTRIA umweltbundesamt

Roadmap of PAKS NPP Lifetime Extension (LTE)

Summary of Current Status and Open Questions of the Bilateral Meetings between Austria and Hungary

lebensministerium.at

ÖSTERREICHISCHES ÖKOLOGIE INSTITUT

ROADMAP OF PAKS NPP LIFETIME EXTENSION (LTE)

Summary of Current Status and Open Questions of the Bilateral Meetings between Austria and Hungary

Oda Becker, Mathias Brettner, Kurt Decker
Helmut Hirsch, Adhipati Y. Indradiningrat

By Order of the
Federal Ministry for Agriculture, Forestry,
Environment and Water Management,
Project Management Department V/6
"Nuclear Coordination"
GZ: BMLFUW-UW.1.1.2/0003-V/6/2011

lebensministerium.at

REPORT
REP-0402

Wien 2012

Project management

Franz Meister, Umweltbundesamt

Authors

Oda Becker, Scientific consultant (Confinement and BDBA; Terror Attacks)
Mathias Brettner, Physikerbüro Bremen (Seismic Design; Ageing Management)
Kurt Decker, University of Vienna, Faculty of Earth Sciences, Geography and Astronomy (Seismic Hazards)
Helmut Hirsch, cervus nuclear consulting (RPV; PU and Fuel Development)
Adhipati Y. Indradiningrat, cervus nuclear consulting
(RPV; PU and Fuel Development)

Layout and typesetting

Elisabeth Riss, Umweltbundesamt

Title photograph

© iStockphoto.com/imagestock

For further information about the publications of the Umweltbundesamt please go to: http://www.umweltbundesamt.at/

Imprint

Owner and Editor: Umweltbundesamt GmbH
Spittelauer Lände 5, 1090 Vienna/Austria

Printed on CO_2-neutral 100% recycled paper

ISBN 978-3-99004-206-9

CONTENT

EXECUTIVE SUMMARY

At the request of Austria, a transboundary Environmental Impact Assessment Procedure pertaining to the Lifetime Extension (LTE) of the Paks Nuclear Power Plant (NPP) started in 2005 and was finalised in 2006.

This procedure was marked by openness, transparency and good neighbourly cooperation. Nevertheless, inter alia due to the early stage of the overall procedure, not all issues could be clarified. Consequently, the Austrian and the Hungarian Delegation agreed that further questions shall be discussed in the framework of the agreement between the Government of Austria and the Government of Hungary on “Issues of Common Interest in the Field of Nuclear Safety and Radiation Protection” (“Bilateral Agreement”) within bilateral consultations.

To this end the Roadmap LTE Paks NPP has been elaborated and finally agreed on in April 2009, specifying, inter alia, a schedule for dealing with certain issues.

This report summarises and evaluates the information received during the Roadmap procedure and gives an overview of open questions and issues to be further addressed.

One part of these issues will be treated within the framework of the project “Stress Tests Follow-up Actions”. This part will comprise the issues which are of particular safety relevance for Austria. In the project, various sources are being evaluated – reports from the different phases of the EU Stress Tests and from the CNS Second Extraordinary Meeting 2012, as well as information from bilateral meetings etc. The project covers the Austrian neighbouring countries with NPPs. After conclusion of the project, these issues will be subject to monitoring, with high priority, in the framework of the respective bilateral agreement.

The other open question/issues are also of interest but of lower priority to Austria and could be further discussed, as appropriate, at the regular bilateral meetings.

Seismic Hazard and Design

Discussions of the issue of seismic hazard assessment during previous Roadmap meetings and information obtained from the Hungarian side during the EU Stress Tests indicate that Hungarian experts took significant efforts to identify and map Quaternary and active faults in the site vicinity and near region of Paks NPP. It is concluded from that information that several capable faults have been identified. However, no information on the further analysis and assessment of these faults (e.g. by paleoseismological methods) has been provided. It is further unclear whether these active faults are adequately considered in the seismic hazard assessment or not.

A systematic assessment of Quaternary faults and the parameterisation of slip history, youngest slip events, fault geometry and slip velocity is of utmost importance for the reliability of seismic hazard assessments. The Austrian side would therefore highly appreciate to get additional information ensuring that the necessary investigations have been performed and their results are adequately integrated into seismic hazard assessment.

Originally, Paks NPP was not designed against seismic loads. A large effort was undertaken to upgrade the plant to the level of the design basis earthquake defined in the course of an updated seismic hazard assessment (peak ground acceleration = 0.25 g).

Concerning the methodologies applied to seismic upgrades, a "mixed" approach was chosen. Its basis were procedures and criteria usually applied to a new design in combination with methods and techniques developed for seismic re-evaluation of operating nuclear power plants. According to the information given by the Hungarian experts the effectiveness of the upgrades was evaluated. The consequences of structural upgrades with respect to the dynamic answers of structures were also assessed. Consequently no open questions remain concerning the methodologies applied to seismic upgrades.

In the process of seismic upgrading the effects of specific measures to be implemented were quantified by different versions of Seismic Probabilistic Safety Assessment (SPSA). The SPSA was therefore systematically used as an important analytical tool, which is a reasonable approach in the course of the implementation of seismic upgrades in Paks NPP.

According to the results on the SPSA, the core damage frequency (CDF) due to seismic events is actually 4.8 x 10^{-5}/a, which is by a factor of 5.4 lower than the value stated in the Environmental Impact Study (EIS 2006). This reflects the effect of the seismic upgrades. According to PSA results presented by the Hungarian side, the actual overall CDF value is below 1.0 x 10^{-4}/a with earthquakes still providing the dominant contribution to the CDF.

As a consequence of the upgrades there is only a low conditional probability for core damage for seismic loads up to the updated design basis. On the other hand, there are no significant reserves for peak ground accelerations (PGA) above this value.

The national report of Hungary prepared in the framework of the EU Stress Tests mentions several seismic issues which are not fully resolved and should be further addressed, but differ in their significance. The biggest issue is the potential for soil liquefaction, as it could act as an important initiator for a common cause failure (CCF) leading to concurrent failure of systems vital for safety.

It is suggested that all issues should be treated in the framework of the above mentioned Project "Stress Tests Follow-up Actions".

Reactor Pressure Vessel

The reactor pressure vessel (RPV), which contains the reactor core, is the central component of a nuclear power plant (NPP). The most important ageing mechanism of the RPV is embrittlement of materials close to the core through neutron radiation. Pressurised thermal shock (PTS) can threaten the integrity of an embrittled RPV.

Most of the questions and points of discussion raised by the Austrian side have been treated and clarified in the document "Summary of PTS calculations" (PTS 2008) provided by the Hungarian side in 2008, such as neutron fluence calculations, selection of PTS initiating events and a comprehensive description of the surveillance system as well as the extension of the surveillance programme at Paks NPP. Results of the neutron fluence calculations and thermal-hydraulic analyses at Paks NPP were also provided in the previously mentioned document. Other issues like the adaptation of ASME boiler and pressure vessel code and neutron flux on RPV wall and critical welding were treated in the discussion at bilateral meetings.

Some issues which are still not yet completely cleared concern the scope of the database of un-irradiated material, the applied safety margin for the ductile-to-brittle-transition temperature in PTS analyses, consideration of dose rate effect caused by copper-rich precipitates and the monitoring method of operational changes with the help of specimens of the surveillance programme at Paks NPP. There is also an open question regarding the detection limit of ultrasonic examinations in relation to the in-service inspection for under-cladding cracks.

These open questions could be further discussed, as appropriate, at the regular bilateral meetings between Hungary and Austria.

Power Uprate and Fuel Development

The power of the units at Paks NPP has been increased to 108% of the original level. The power uprate at Paks NPP is connected to fuel development to reach the targeted power level (first phase of fuel development) and to achieve a more economical fuel cycle after the power uprate (second phase of fuel development). In the second phase, the new fuel has an average uranium enrichment of 4.2%. The use of this fuel has been tested in unit 4. Test results of the new fuel have been presented.

Due to the increase of the reactor thermal power, the risk of plant operation can be increased. Margins relevant for safety might be reduced and plant ageing is accelerated. These issues were also dealt with in the Roadmap LTE Paks NPP. Effects of the power uprate on lifetime extension (LTE) were discussed.

In the course of the Roadmap, many questions related to power uprate and fuel development have been clarified – a significant amount of information regarding the new fuel, safety factors for reactor operation, changes of safety systems and the result of accident analyses has been provided. However, there are still some points which have to be clarified.

Quantitative information on the time frame until overheating in the case of SBLOCA without HP injection and available time frames for successful operator actions in other cases has not been provided. Further explanation on the evaluation of severe accident management (SAM) mitigative actions would still be welcome.

All above mentioned open issues will be dealt with in the framework of the project “Stress Tests Follow-up Actions”, topic 3.

Additionally, it would be appreciated to receive information on the results of the programme to track and review the trending of parameters after PU within the regular bilateral meetings.

Confinement System and BDBA

The confinement system of VVER-440/213 consists of a system of rooms, containing the primary circuit, the bubble condenser tower with large trays filled with water and air traps and an active spray system. The behaviour of the confinement system is of crucial importance for all severe accidents. The amount of radioactive releases is determined by the confinement's leak-tightness and its capability to withstand loads more severe than the design basis. Comprehensive information about the structure of the containment, the design data, the leakage rate and its capacity under accident conditions were given by the Hungarian side.

As a pre-condition for the planned LTE the nuclear authority required that the modifications necessary for the management of beyond design basis events and severe accidents shall be completed (HAEA 2011). All severe accident management (SAM) modifications planned before the Stress Tests have already been implemented at unit 1 and will be implemented at units 2 to 4 until December 31, 2014. The most important SAM issue is the external cooling of the reactor pressure vessel (RPV) by flooding the reactor cavity to prevent RPV failure. According to HAEA (2011) the calculations of the so-called in-vessel retention (IVR) concept were justified in the frame of CERES experimental analyses. Because of its importance, it might be desirable that the issue of this IVR concept should be taken up again, in particular results from the CERES tests.

However, during the slow increase of pressure caused by steam produced during the external cooling of RPV, the unfiltered release through the stack could be necessary to avoid containment failure. Thus, an active containment cooling system to prevent over-pressurisation of the containment will be designed and installed in the next phase of the accident management modifications (final deadline December 15, 2018). The installation of a filtered venting system is not planned.

The ENSREG Peer Review Team concluded that the Hungarian approach to manage severe accidents seems to be comprehensive; no major weak points for the severe accident management were identified. Nevertheless, there are areas where further improvement may be achieved. Several improvements of the SAM, particularly regarding the management of accidents in the spent fuel pools and multi-unit accidents, are envisaged.

Because of the importance of the containment capability as well as the SAM, some further information on this issue would be of interest to the Austrian side.

It is planned to further discuss the open issues in the framework of the project "Stress Tests Follow-up Actions", topic 3.

Ageing Management

In Hungary several regulatory guidelines for ageing management and in-service inspection have been implemented. These requirements define the basic scope of the ageing management programme at Paks NPP. Full implementation of the ageing management programme should be accomplished.

Based on the available information we conclude that a comprehensive and systematic approach for ageing management has been implemented in Paks NPP – at least this applies to mechanical components, as no further information concerning ageing management of structures and I&C components has been presented. Part of the ageing management programme is the database/expert system DACAAM. Based on the available information it seems to be well suited for this purpose.

With the exception of activities concerning steam generator corrosion no detailed information is available with respect to the experiences concerning the ageing management programme at Paks NPP. Aspects concerning these experiences are e.g. the efficiency of the coordination and cooperation of the different departments responsible for certain aspects of ageing management as well as recent leakage events.

According to our understanding the approval of the adoption of ASME Code section XI to Paks NPP by HAEA is still under way. The adoption necessitates a post evaluation of materials, design and operation of the relevant components. Up to now no detailed information about the approach for ASME code adoption and the envisaged doubling of in-service inspection (ISI) cycle length has been presented.

The remaining open issues could be discussed, as appropriate, within the regular bilateral meetings.

Terror Attacks

Only very general information was provided which is far from being sufficient to disprove that large radioactive releases are possible after a terror attack. Indeed, these hazards exist for all commercial nuclear power plants. In addition, there seem to be some specific vulnerabilities at VVER-440/213 plants. An important weakness appears to be that there is no protection against an aircraft crash at Paks NPP. This would also imply high vulnerability against other modes of attacks from the outside. Further information regarding the issue of terror attacks and the design basis threat (DBT) would be of great interest to the Austrian side, considering the large consequences of a potential attack. Vulnerabilities, attack scenarios and potential consequences can and should be discussed in an appropriate general manner, and in an appropriate setting. Due to the sensitivity of the topic, discussion would require an appropriate framework.

1 OPEN QUESTIONS/ISSUES TO BE FURTHER ADDRESSED

Seismic Issues

We recommend that the following issues should be further addressed in the bilateral process between Hungary and Austria. It is suggested that the listed issues are treated in the framework of the project "Stress Tests Follow-up Actions" (topic 1):

- Reflection seismic data acquired during the seismic hazard assessment programmes in the late 1990ies and early 2000nds identified several Quaternary faults in the vicinity and near-region of the site. Have these faults been investigated with proper methodologies in order to constrain slip histories, youngest slip events, fault geometries and slip velocities?
- The information provided for the EU Stress Tests mentions that some paleoseismological investigations have been carried out. Have these methods been applied in a systematic way to analyse the faults in the vicinity and near-region of the site and what are the results of these studies?
- SHA apparently includes earthquake recurrence models derived from fault parameters such as fault dimension and slip rate (models by Ove Arup). The Austrian experts would highly appreciate to get more detailed information on this issue. Have these models been applied to those Quaternary faults, which were identified by reflection seismic? What are the assumptions and input parameters for the fault models? Does the currently valid PSHA account for Ove Arup's modelling results?
- Current PSHA includes a logic tree approach, which attributes 10% probability to a model including active faults and 90% probability to "no faults". What is the justification for attributing such low probability to the active fault branch of the logic tree at the background of the existing evidence for Quaternary faults?
- The results of the assessments concerning necessity of measures envisaged to increase robustness of the plants against earthquakes – as presented in chapter 2.2.4. of the "National Report of Hungary on the Targeted Safety Reassessment of Paks Nuclear Power Plant" (HAEA 2011) – and their respective implementation according to the "National Action Plan of Hungary on the implementation actions decided upon the lessons learned from the Fukushima Daiichi accident" (HAEA 2012), especially:
  - The results of further assessments concerning the potential impact of soil liquefaction and the respective implementation of additional measures to avoid CCF failure of vital safety functions. The Austrian side would highly appreciate to get information on the type of the safety relevant structures, systems and components endangered by liquefaction and on the measures envisaged to strengthen these SSCs.
  - The consequences of a potential failure of the three common demineralised water storage tanks of Installation II due to damages at the service building with respect to (secondary side) decay heat removal after an earthquake.
  - The results of the re-evaluation of the question seismic shutdown in the frame of the reconstruction project of the seismic instrumentation.
  - The potential for site flooding due to failure of pipelines of the main condenser cooling water system.

- The necessity to perform inspections of already installed anchor bolts to check whether they have been mounted correctly.
- Results of analyses concerning the possible consequences of a superposition of operating conditions during low-power and shutdown operation of short duration with a design basis earthquake.
- It should be clarified whether the deterministic safety case relies on successful operator actions within short time periods after an earthquake (e.g. within 30 minutes).

Reactor Pressure Vessel (RPV)

The following questions are of interest to Austria and could be further discussed, as appropriate, at the regular bilateral meetings:

- Regarding the database of un-irradiated material, what is the scope of the recording of properties?
- Does the difference of 25 °C (mentioned in PTS (2008) Chapter 4.1.1) represent the safety margin for T_k in the PTS analyses?
- Has consideration been given to a dose rate effect caused by copper-rich precipitates (CRP)?
- Regarding the ISI programme, what is the limit of detection for ultrasonic (UT) examinations for under-cladding cracks (smallest crack depth which can be detected with certainty)? Do the 6 mm mentioned represent the detection limit?
- Please provide more elaboration on the method for monitoring operational changes by using specimens of the surveillance programme.

Power Uprate and Fuel Development

Of the following questions/issues, all but the last one will be treated in the framework of the above mentioned project “Stress Test Follow-up Actions”, topic 3. The last issue could be discussed, as appropriate, at the regular bilateral meetings.

- In case of SBLOCA without successful HP injection, how long is the time until overheating for the power level after PU and for previous power level?
- Quantitative information on available time frames for successful operator actions for other cases (both for 100% and 108% power level) would still be of interest to the Austrian side.
- Quantitative information on the changes in the integrated mass of produced H_2 and the production rate after PU would still be welcome.
- More information on the evaluation of SAM mitigative actions in relation to PU would provide better understanding.
- Information on the results of the programme to track and review the trending of parameters after PU would be appreciated.

Confinement and BDBA

Of the following questions/issues, all but the first one will be treated in the framework of the above mentioned Project "Stress Tests Follow-up Actions", topic 3. The first issue could be discussed, as appropriate, at the regular bilateral meetings.

- The design leakage rate is 14.7 vol% per day for the maximum DBA case (LBLOCA). A full pressure test for unit 3 showed a lower leakage rate (4.8 vol% per day). The results of full pressure tests at the units 1, 2 and 4 would still be of interest to the Austrian side.
- Accidents with the steam generator (SG) tube or collector rupture lead to particularly high releases, since the containment is bypassed. Measures are planned or already implemented at Paks; due to time constraints, this topic could not be addressed. The clarification of this issue would still be welcome.
- Because of its importance, the in-vessel retention (IVR) concept should be taken up again in case the Austrian experts identify remaining open questions concerning the results from the CERES tests. The implementation of IVR concept at all four units in Paks should be monitored in any case.
- For the Austrian side information on SA source terms and large release frequencies after implementation of SAM strategy and mitigative actions is of high relevance.
- As a result of the Stress Tests several improvements of SAM, particularly regarding the management of accidents in the spent fuel pools and multi-unit accidents, are envisaged. Information about the planned measures and results of studies would be of great interest to the Austrian side, especially regarding:
  - the active containment cooling system aiming at the prevention of the slow over-pressurisation of the containment,
  - the water supply to the spent fuel pool from an external source.

Ageing Management

We recommend that the following issues should be further addressed as appropriate in the bilateral process between Hungary and Austria in the framework of regular meetings:

- The ageing management programme for structures and I&C components.
- The experiences with respect to the general performance of the ageing management programme in Paks NPP. Aspects concerning these experiences are e.g. the efficiency of the co-ordination and cooperation between the different departments responsible for certain aspects of ageing management as well as recent leakage events (steam generator drainage pipe and a water purification system pipe in unit 4).
- The adoption of ASME Code section XI. Aspects concerning this adoption are a post evaluation of materials, design and operation of the relevant components.
- The technical justification for the doubling of in-service inspection (ISI) cycle length.

Terror Attack

Further information regarding the issue of terror attacks (e.g. DBT) would be of great interest to the Austrian side, considering the large consequences of potential attack. Vulnerabilities, attack scenarios and potential consequences can and should be discussed in an appropriate general manner. Due to the sensitivity of the topic, discussion would require an appropriate framework.

2 INTRODUCTION

At the request of Austria, a transboundary Environmental Impact Assessment Procedure (according to the EIA Directive and the Espoo Convention respectively) pertaining to the Lifetime Extension (LTE) of the Paks Nuclear Power Plant started in 2005 and was finalised in 2006.

This procedure was marked by openness, transparency and good neighbourly cooperation. Nevertheless, inter alia due to the early stage of the overall procedure, not all issues could be clarified.

Consequently, at a consultation which was held at the Ministry of Environment and Water of the Republic of Hungary in Budapest on July 10, 2006, the Austrian and the Hungarian Delegation

- agreed that further questions which might be raised by the Austrian side shall be discussed in the framework of the Agreement between the Government of Austria and the Government of Hungary on Issues of Common Interest in the Field of Nuclear Safety and Radiation Protection ("Bilateral Agreement");
- agreed to continue bilateral consultations on issues regarding severe accidents, in the framework of the "Bilateral Agreement";
- agreed that the Bilateral Commission, established to implement Art. 12 of the "Bilateral Agreement", might hold extraordinary meetings – on either party's request – to deal with these issues, as well as with issues which would come up later in this context, as appropriate;
- recalled that Art. 10 of the "Bilateral Agreement" foresee that comments by one party shall be transmitted to and considered by the competent authority of the other party.

The Roadmap LTE Paks NPP was finally agreed on in April 2009 after extensive bilateral consultations and was attached to the minutes of the regular 14th bilateral meeting under the Bilateral Agreement mentioned above.

Specific questions relating to six technical issues – Seismic Issues; RPV; Power Uprate & Fuel Development; Confinement & BDBA; Ageing Management and Terror Attacks – should be discussed during the Roadmap process between 2008 and 2012.

Discussion started at the 14th bilateral meeting (November 25, 2008, Vienna) with the following issues:

- Confinement & BDBA: Confinement behaviour during DBA, BDBA: Time pressure curves, leak rates as a function of pressure, additional information on BDBA source terms
- Reactor pressure vessel (RPV): Database for unirradiated material, surveillance programme, dose rate effect; further information on PTS analyses (assumptions, methodology, results, consequences)

For the 15th bilateral meeting (November 30, 2009, Budapest) the following issues were intended for discussion:

- Ageing Management: Development of ageing management and in-service-inspections (particularly concerning steam generators); steam generator corrosion.

- Confinement & BDBA: Influence of power uprate and fuel development on intervention time and source term; Severe accident management measures planned and to be implemented (technical & organisational).
- Terror Attacks: Vulnerability of the plant to a spectrum of possible attacks, Hungarian regulations concerning design basis threat.

For the 16th bilateral meeting (November 22/23, 2010, Schloss Hernstein) the Roadmap schedule defined:

- Seismic Issues: New investigations of seismic issues, new assessment of seismic hazard; further seismic backfitting activities.
- Power Uprate & Fuel Development: Experiences from power up-rate and 1st phase of fuel development, including all systems and components.

At the 17th bilateral meeting (November 8, 2011, Pécs) only one issue was discussed:

- Power Uprate & Fuel Development: Status of 2nd phase of fuel development (including discussion of effects of 2nd phase of fuel development on neutron fluence in RPV wall).

As scheduled the Roadmap LTE Paks NPP was finalised at the 18th bilateral meeting (December 6/7, 2012, Eisenstadt).

In the course of the Roadmap LTE Paks NPP, many questions have been clarified and a significant amount of information regarding the six technical issues has been provided. The procedure was marked throughout by openness, transparency and good neighbourly cooperation. The Hungarian side provided detailed papers and presentations to the specific topics; these documents are listed in Annex 2 of this report.

However, there are still some points which have to be clarified and some new questions have been raised in the framework of the European Stress Tests.

This report summarises and evaluates the information received during the Roadmap procedure. Furthermore, an overview of "open questions/issues to be further addressed" has been compiled.

One part of these issues will be treated within the framework of the project "Stress Tests Follow-up Actions". This part will comprise the issues which are of particular safety relevance for Austria. In the project, various sources are being evaluated – reports from the different phases of the EU Stress Tests and from the CNS Second Extraordinary Meeting 2012, as well as information from bilateral meetings etc. The project covers the Austrian neighbouring countries with NPPs. The project "Stress Tests Follow-up Actions" mainly focuses on the three topics of the Stress Tests ((1) natural hazards, (2) loss of safety functions and (3) severe accident management); however, issues from other topics also can be included, in exceptional cases. For each of the issues which have been selected for this project, a technical justification will be compiled. After conclusion of the project, these issues will be subject to monitoring, with high priority, in the framework of the respective bilateral agreement.

The other open questions/issues also are of interest but of lower priority to Austria, and could be further discussed, as appropriate, at the regular bilateral meetings. Due to the sensitivity of the topic "terror attacks", the discussion concerning this matter would require an appropriate framework.

3 SEISMIC HAZARD AND DESIGN

The recent earthquakes in Japan (Tokohu and Niigataken Chuetsu-Oki) and the respective consequences for the nuclear power plants concerned (e.g. Fukushima Daiichi and Kashiwazaki-Kariwa) have highlighted the importance of a robust design against earthquakes. With respect to Paks NPP it is of special interest that the plant practically was not designed against seismic loads at all. In the course of an updated seismic hazard assessment a peak ground acceleration (PGA) of 0.25 g was defined for the design basis earthquake (DBE). A large effort was undertaken to upgrade the plant to the level of the DBE.

These upgrades and the core damage frequency (CDF) due to seismic events have been part of the discussions concerning the environmental impact assessment of the lifetime extension for Paks NPP right from the beginning. In a report to the Austrian Government on Paks NPP Lifetime Extension (UMWELTBUNDESAMT 2006) the Austrian experts stated that earthquakes can lead to severe damage of a nuclear power plant, if the plant is not properly designed against the seismic loads. A core melt accident could result, possibly with damage to the containment and large early releases. It was judged that seismic events are regarded as an important potential contributor to NPP risk worldwide.

3.1 Summary of information provided

Seismic Hazard

The seismic issue was addressed by a dedicated presentation at the 16th bilateral meeting in 2010 (ELTER 2010). During this meeting a number of questions on technical details arose, which turned out to be beyond the scope of the bilateral meeting. These questions were summarised by the Umweltbundesamt and forwarded to the Hungarian side (UMWELTBUNDESAMT 2010). The Hungarian side provided written response to the outlined questions timely before the 17th bilateral meeting in 2011 (KATONA & BAREITH 2011).

During the preparation of the agenda for the 17th bilateral meeting both, the Hungarian and Austrian side, decided not to enter a detailed discussion on seismic issues at this meeting as the Hungarian experts were expected to provide additional information on the seismic safety of the Paks NPP in the framework of the European Stress Tests. These data should be used as a basis of the further bilateral discussion.

Relevant and valuable data for the discussion of seismic issues in the Roadmap LTE Paks NPP is included in the Hungarian National Report submitted to the European Stress Tests (HAEA 2011), the dedicated presentations given during the Stress Tests (KATONA 2012; RÓNAKY 2012) and the ENSREG Peer Review Country Report (ENSREG 2012).

The National Stress Tests report (HAEA 2011) includes a detailed demonstration of the site-specific seismic hazard and the plant's seismic safety as stipulated by the Stress Tests procedure. The report was discussed and reviewed in the frame of the Stress Tests´ Topic 1 "Initiating events" (earthquakes, flooding and extreme weather). During this review supplementary data and information was provided in the formats of written answers to questions arising from the National Report and dedicated presentations by the Hungarian regulator held at the Peer Review meeting at Luxembourg.

This material is accessible to the Austrian expert group and may be used in the Roadmap LTE Paks NPP discussion by courtesy of the Hungarian regulator. The Head of the Hungarian delegation to the European Stress Tests, Dr. József Rónaky, suspended the confidentiality of the material for its use in the Roadmap and bilateral process upon oral Austrian request.

New investigations and seismic hazard assessment performed for LTE

The Austrian side asked for information on what kinds of investigations and new assessments of seismic hazard were performed in the framework of the LTE (Life Time Extension) process for Paks. Measures undertaken after the reinforcement and qualification programme, which has been implemented during the time period from 1993 to 2003, should be identified. The question intended to clarify whether the license extension is based on new and additional hazard assessments or not.

The Hungarian side informed that Seismic Hazard Assessment (SHA) for the Paks site was performed using Probabilistic Seismic Hazard Assessment (PSHA) in accordance with IAEA safety standards and international practice. As a result, in 1996 the design base earthquake level of PGA_H=0.25 g and PGA_V=0.20 g for the occurrence probability of 10^{-4}/year was established. Those values, corresponding to the SL-2 level (Seismic Level 2 corresponding to the highest safety level), have not changed in the subsequent re-assessments and are still relevant today.

Data and methods used for earlier PSHA have been re-evaluated in 2007 and 2008. This study also included an analysis of the records of the continued microseismic monitoring of the site and an assessment whether these data require modifying the seimotectonic model or not. The study was apparently carried out during the 2nd PSR (Periodic Safety Review) of the plant (KATONA & BAREITH 2011; KATONA 2012). The results of the study were published in TÓTH et al. (2009).

New investigations further included a sensitivity study to highlight the type of input data that dominates the uncertainty of the PSHA results. The Austrian side was further informed that scopes and methods for an update of PSHA have been defined in a post-PSR action in 2008.

In 2011, a dedicated study of liquefaction hazard was performed by Hungarian experts in cooperation with experts from the Technical University of Berlin. The results of that quantitative assessment were presented during the EU Stress Tests. Accordingly, the safety margins against soil liquefaction reveal only narrow margins for the sediment layers between 10 and 20 m beneath the site leading to the conclusion that liquefaction is expected as a dominating damage mode for seismic accelerations exceeding PGA_H=0.25g. Soil liquefaction and consequent building settlement is expected to have major effects on inter-building connections.

Assessment of Quaternary faults in the near-region of the site

A group of questions addressed the use of geological and microseismic data in the existing PSHA (Probabilistic Seismic Hazard Analysis). The information request particularly intended to clarify whether or not a serious attempt was made to identify active faults in the near-region of the NPP and to assess the seismic capability of known Quaternary faults, e.g., by paleoseismological studies.

The topic arose from recent publications showing evidence for Quaternary and active faults in the near-region of the plant (e.g. TÓTH 2003; HORVATH & BADA 2004; MAGYARI et al. 2011), which appeared to be not properly integrated in the earlier seismic hazard assessments. The topic has been regarded as a highly important issue as the validity of the current SHA strongly depends on the correct assessment of the near-regional active faults.

Response to this key information request of the Austrian side was received before the 17th bilateral meeting in 2011 (KATONA & BAREITH 2011). As the information supplied in this document was not regarded to be sufficiently clear it was decided to track the issue during the EU Stress Tests. In fact the Hungarian side used the Stress Tests as an opportunity to respond to the open questions dedicating a large part of the Country Presentation at the Stress Tests Peer Review in Luxembourg to topics raised by the Austrian experts. This approach and the privileged treatment of the Austrian concerns are highly appreciated. Information obtained through the EU Stress Tests is included in HAEA (2011), ENSREG (2012), KATONA (2012), RÓNAKY (2012).

The Hungarian side informed that faults in the site vicinity and the near-region of the NPP were investigated by shallow reflection seismic profiling, paleoseismological investigations and microseismic monitoring. It was concluded that the recorded microseismicity does not highlight active faults (ELTER 2010). It was stated that the source zone models used in PSHA accounts for the microseismic data. Stress Tests presentations further claim that paleoseismological investigations have been performed without providing details on the results of such studies. The only reference to paleoseismological investigations is included in (KATONA 2012) referring to work by Árpád Magyari in the region of Bicske, which, however, is located some 100 km from the NPP site. It appears that no such analyses are available for faults close to the site.

The presentation during the Stress Tests Peer Review meeting at Luxembourg further indicates that active faults have been modelled in the existing PSHA by a logic tree approach. In that PSHA a 10% probability has been assigned to local fault sources whereas a 90% probability has been selected for seismotectonic scenarios without active faults (KATONA 2012).

Hazard assessment apparently also includes an approach to model earthquake recurrence intervals for a number of active faults. These fault models seem to consider input parameters such as fault size and fault slip rate. It appears that these models were prepared by the engineering company Ove Arup. No additional details on the type of model, model assumptions and input data are provided in KATONA (2012). The company Ove Arup apparently only contributed to the seismic hazard assessments prepared between 1993 and 1994, which arrived at higher ground motion values (0.35 g) than the current SL-2 level. It is unclear whether these fault models or other approaches for modelling active faults in terms of seismic hazard are included in the currently valid PSHA or not.

Seismic hazard assessments for siting and licensing of Paks 5&6

Requested information further addressed the identification of the measures (hazard reviews, new full-scope PSHA etc.) required for the siting and licensing of Paks 5&6 and the possible impact of the results of such new assessments on the existing units.

Measures for seismic hazard assessments required for the siting and licensing of Paks 5&6 are sketched in KATONA & BAREITH (2011). According to Volume 1 of the Hungarian regulation, siting requires a full scope site investigation and an individual site license. Evaluation requirements are defined in Volume 7 of the Nuclear Safety Regulation, which complies with IAEA safety requirements (IAEA 2003b). Accordingly, new assessments will have to use state-of-the-art methodology accounting for international practice and IAEA safety guidelines (IAEA 2010). Any such process will build on the existing database and experience obtained from previous SHA.

It is further stated that the licensee of the existing plant is obliged by the Act on Atomic Energy (CXVI, 1996) to review the plant’s safety in the light of new scientific evidences and take measures if needed. New evidence obtained during siting and licensing for Paks 5&6 therefore have to be considered for the operating plant as well. The Hungarian side explains that for the existing plant a remaining operational time of 20 years will be considered in seismic risk assessment.

Seismic Design

The Preliminary Environmental Study (2004) briefly mentioned that seismic upgrades of building structures and safety systems had been performed, but there was no systematic discussion of seismic design issues. Furthermore, it was mentioned that instability of the ground around unit 4 of the Paks NPP could lead to damages to buildings. It was noted that stabilisation of the ground through injections might already become necessary during the first 30 years of operation (PES 2004).

The Environmental Impact Study (2006) showed that seismic events contributed to the overall core damage frequency (CDF) of 3.0×10^{-4}/a and unit with 86%. The EIS stated that measures for risk reduction concentrated on seismic upgrades and that measures were already under way (EIS 2006). The Austrian side judged that the value for the core damage frequency contribution of seismic events given in the EIS was significantly higher than the target value for CDF for existing nuclear power plants which has been formulated by the International Nuclear Safety Advisory Group of the IAEA (1×10^{-4}/a). It was concluded that a comprehensive and up-to-date picture of the seismic hazards associated with operation of Paks NPP was desirable including information on the state of the upgrades and quantitative information on the reduction achieved by upgrades up to that time and to be achieved in the future.

Additional information was provided by the Hungarian side during the discussion at a public hearing on June 6, 2006. An estimate was given for the reduction of CDF due to seismic upgrades implemented until that time. A value of 6.6×10^{-5}/a was quoted at the hearing.

Based on the information available up to June 2006 it was concluded in a report to the Austrian Government (UMWELTBUNDESAMT 2006), that seismic events were still the dominant contributors to CDF at Paks. Therefore more detailed information on the state of the upgrades and the methodology and results of the latest seismic risk analyses would be of considerable interest. Forthcoming new investigations of seismic issues were judged to be of interest from the Austrian point of view and should be closely followed. Also the issue of additional seismic upgrades should be followed further.

Starting from this state of knowledge further information concerning the re-assessment of seismic hazards and seismic back fitting activities were presented by the Hungarian side during the 16th bilateral meeting in 2010 (ELTER 2010). Concerning this information some open questions remained especially concerning the methodologies applied to seismic upgrades and the definition of the screening criteria for equipment applied in the context of the Seismic PSA (SPSA). Based on information contained in the presentation and in openly accessible literature a list of further questions was compiled by the Austrian side and transferred to the Hungarian side. These questions have been answered by Hungarian experts in KATONA & BAREITH (2011). Additional information is provided in the "National Report of Hungary on the Targeted Safety Re-assessment of Paks Nuclear Power Plant" (HAEA 2011) prepared in the context of the EU Stress Tests for nuclear power plants.

The information contained in these documents and in openly accessible literature permits an overview of the activities concerning the reconstitution of the seismic design basis and seismic upgrades for Paks NPP. For that reason a description of these activities has been worked out for this report.

In the following, the available information regarding the reconstitution of the seismic design basis and seismic upgrades for Paks NPP is summarised.

Design basis and chronology of seismic upgrades

As a consequence of the recognised seismic hazards and parallel to their evaluation a seismic upgrade programme for Paks NPP was launched. A chronology of the re-evaluation and the upgrades has been provided in ELTER (2010):

1986:	Recognised: the seismic hazard was dramatically underestimated in the design
1986–1990:	first phase of seismic hazard re-evaluation and associated site geological, geophysical, seismological investigations
1990–1996:	full scope site investigations in accordance with IAEA Safety Series 50-SG-S1
1992–1995:	easy-fix programme
1996:	new design base earthquake, soil liquefaction study, capable fault study
1996–2002:	seismic safety programme, re-design and implementation of upgrades
2000–2002:	update and extension of SHA for seismic PSA

2003: seismic PSA and additional upgrades

2007: 2nd Periodic Safety Review

2008: re-evaluating of PSHA and sensitivity study

2009 - IAEA Extra-budgetary Project on "Seismic Safety of Existing NPPs"; International Seismic Safety Centre

Methodologies for seismic upgrades

The "classical" deterministic seismic design is based on the application of relevant codes and standards (e.g. Eurocode, ASME, KTA). This kind of design is usually conservative (e.g. due to restriction to linear-elastic methods, low damping values, limited consideration of plastic energy absorption, lower bound values for specified material's parameters). Therefore systems, structures and components (SSC) typically contain inherent margins which allow them to accommodate higher loads than the design loads until failure occurs – their seismic capacities (usually expressed as PGA values) are higher than the design values. This provides some kind of reserve against seismic loads.

In the case that a re-evaluation of seismic hazards results in stronger seismic impacts than those that have been accounted for in the design basis, a sufficient robustness of the already built NPP against the enhanced loads has to be shown. Special methods have been developed for this purpose (mainly in the US). A basic explanation of methods for the seismic (re-) evaluation of existing NPPs is contained in IAEA (2003a) and IAEA (2009).

Starting point of the methodologies for the seismic re-evaluations of existing NPPs are the above mentioned inherent reserves contained in the design of the NPP. Based on the actual seismic design it is evaluated to what extent higher loads can be accommodated due to the existing reserves, possibly after some modifications to enhance the seismic robustness of certain SSCs. Often one wants to show that seismic loads connected to a defined earthquake stronger than the design basis earthquake (DBE), the so-called "review-level-earthquake" (RLE), can be accommodated. The main principles and problems encountered in the context of seismic re-evaluations during the 1990's were expressed by the IAEA in the following way (GODOY & GÚRPINAR 2001):

"Special considerations arise when the nuclear power plant has already been constructed and is in operation. Seismic qualification is distinguished from seismic re-evaluation primarily in that seismic qualification is intended to be performed at the plant design stage, whereas seismic re-evaluation is intended to be conducted after the plant has been constructed. For those purposes the following considerations are relevant:

(1) It is a known technical finding that industrial facilities, especially NPPs, which have been sited, designed and constructed using good engineering practice and internationally accepted regulations have an inherent capability to resist earthquakes larger than the earthquake used in their original design. This inherent capability is a direct consequence of the conservatism that exists in the seismic design and is usually described in terms of "seismic design margin ".

(2) *At the design stage it may be easy to add certain seismic design margins in traditional ways because the associated costs are relatively low. Typically, seismic design criteria applicable to NPPs are specified in such a way that, although it is known that they introduce very large seismic design margins, their size is not usually quantified. Because of the ways that seismic design margin is introduced by design criteria, seismic margin typically varies greatly from one location in the plant to another, from one structure, system and component to another, and from one location to another in the same structure.*

(3) *After the plant is constructed, however, it may be very costly to add the same seismic design margin if it is done in the traditional ways used during the design stage. At the postconstruction stage, an adequate margin can be ensured through the use of special safety evaluation procedures. These procedures are aimed in raising more efficiently only the lower and most safety significant margins than do traditional seismic design criteria and methods. Nevertheless, although there may be special difficulties in performing hardware modifications during the operation period of an existing plant, the significance of these difficulties cannot be judged until the plant's capability to withstand earthquakes is systematically determined.*

(4) *Neither the IAEA, nor any regulatory authority, has established definitive and comprehensive guidelines for the seismic re-evaluation of existing operating nuclear power plants. Although some guidelines do exist for the seismic re-evaluation of existing nuclear power plants built to earlier standards, these are not established at the level of a regulatory guide or its equivalent. Nevertheless, a number of existing nuclear power plants throughout the world have been and are being subjected to review of their seismic safety. Rational criteria for resolving the main issues were developed, particularly in the USA, which have been adapted for the specific conditions in Western and Eastern European countries.*

(5) *It is also recognized that re-evaluation programmes at existing operating plants are unique and, therefore, plant-specific or regulatory-specific. This means that specific requirements and guidelines have to be developed for each case. The fact that the plant is already constructed and the specific construction details and its 'as-is' conditions can be inspected are also important factors in deciding on the level of effort and methods that can be used in its seismic re-evaluation. In deciding this, it is important to determine whether the plant has (or has not) been originally designed for seismic loads. For instance, in the specific case of the Armenian NPP seismic re-evaluation, this plant presents a good 'reference basis' since it was explicitly designed against earthquakes according to the rules valid at that time in the former USSR."*

All methods for the seismic (re-)evaluation of existing NPPs need quantitative information about the robustness of SSCs against seismic loads. Therefore one important step is the evaluation of the seismic capacity of relevant components and structures. According to IAEA (2003a) the seismic margin capacities should be more conservatively assessed than in case of a conventional seismic evaluation of industrial facilities but less conservative than required for qualification of new NPPs.

The usual starting point for the application of methods to evaluate seismic margins is the basic design of the NPP against seismic loads. The situation concerning seismic re-evaluation of Paks NPP was different and more complicated as the plant was in fact not designed against seismic loads at all. Consequently, there was a large discrepancy between the original seismic design basis (0.025 to 0.05 g) and the actual design basis earthquake (0.25 g horizontal direction). Therefore, as it is pointed out in KATONA & BAREITH (2011), the task was not to re-qualify a plant designed for a certain design basis earthquake for higher seismic loads (which possibly could have been done mainly on basis of seismic margin assessment (SMA) methodology). Instead it was necessary to establish a design basis for a plant already in operation that was not designed against seismic loads. As a consequence, there were two boundary conditions for the process of seismic re-evaluation of Paks NPP:

- Compared to the design phase of a NPP there were limited possibilities to adjust the design to the seismic loads.
- There was no established basis for the application of methodologies to evaluate seismic margins due to the non existing design basis.

These boundary conditions had certain implications with respect to the methodologies applicable for the whole process. On the one hand, due to the missing basic design and the large discrepancies between the original and the actual seismic hazard a re-qualification solely on the basis of the methods for evaluating safety margins was not possible. In this respect the case of Paks NPP was not a typical application of IAEA documents on seismic re-evaluation and upgrading of existing NPPs (KATONA & BAREITH 2011).

On the other hand according to KATONA (2001) and KOTONA & BAREITH (2011) it was recognised at the very beginning of the seismic safety programme that a consequent and full scope re-design in line with design codes and standards and subsequent upgrading might also be impossible. The reason was that this could lead to heavy upgrades and feasibility problems. Therefore, according to the international practice and IAEA recommendations provided in the frame of an IAEA technical cooperation project, the Hungarian authorities allowed the use of methodologies for seismic re-evaluation and re-qualification of operating NPPs (SMA, SQUG/GIP). These are less conservative than the procedures for a new design based on relevant codes and standards.

Consequently, a methodology for the seismic re-evaluation and upgrading of Paks NPP had to be developed which required specific considerations and solutions as

- evaluation and upgrading philosophy and basic requirements for a plant not designed against seismic loads,
- applicability of the re-evaluation and design methods developed in western countries for a nuclear power plant designed and built according to Soviet standards,
- development of a rational approach for upgrading including feasibility aspects.

Eventually for Paks NPP a kind of “mixed” approach was utilised as described in KATONA (2001) and KATONA & BAREITH (2011). The re-qualification was performed by applying procedures and criteria usually applied to a new design in combination with methods and techniques developed for seismic re-evaluation of operating nuclear power plants. The applicability of the re-evaluation methods was carefully studied on the basis of systematic evaluation and comparison of US, German and Soviet design requirements and procedures (a KTA / ASME comparative study was also made) according to KATONA (2001). While it was allowed to follow the liberal approaches for the evaluation of the capacity of existing structures, systems and equipment, the design of upgrading measures had to be performed according to procedures ensuring code compliance. The selection and use of methodologies was graded in correspondence with the safety relevance of the SSCs.

The methods for the evaluation of the as-built seismic capacities of passive load bearing and pressure retaining structures and components were selected in accordance with their Safety and Seismic Classes in the following way (KATONA 2001, KATONA & BAREITH 2011):

- For Safety Class 1 and 2 mechanical components (piping etc.) and Safety Class 2 buildings straightforward design procedures were applied (codes and standards, e.g. for pressure retaining boundaries class 1 and 2 KTA and class 3 ASME; evaluation on the basis of a purely elastic approach). Especially the capacity evaluation of the primary system has been performed in compliance with KTA standards while other piping and equipment in the confinement have been evaluated using also KTA standards but with certain realistic assumptions on the damping and ductility.
- Safety Class 3 structures and components were generally evaluated using realistic assumptions for damping and ductility taken from CDFM SMA (CDFM: Conservative Deterministic Failure Margin) procedure.
- The piping and equipment outside of confinement were evaluated according to SMA type methodologies. Low energy pipes with small diameter and cold pipes with large diameter have been evaluated using simplified calculations and walk-down based methods.
- The design of fixes for all Classes was made per design codes and standards, conservative floor response spectra were used for all Classes.

The qualification of active components was done by several methods according to KATONA & BAREITH (2011):

- in case of replaced or reconstructed systems an equipment qualification and certification of functionality were made by the supplier for the defined floor response spectra; e.g. this was the case for the reactor protection system (Siemens Teleperm XS);
- shaking table testing of sample items;
- qualification via empirical procedures (GIP, GIP-VVER).

For example, relays were qualified by replacing those that could not to be qualified by new ones, by shaking table testing of others and by experience based methods, where they were applicable (PHARE Project).

A brief comparison between the methods applied to Seismic Margin Assessment (SMA), Seismic PSA (SPSA), the design basis reconstitution for Paks NPP and the SPSA for Paks NPP has been provided in KATONA (2011). It summarises the information given above in table form.

Seismic upgrades of SSCs

A main boundary condition for the seismic upgrade of Paks NPP was the basic construction of the plant. While the massive lower part of the reactor building possesses a high robustness against seismic loads the situation was quite different for the gallery building and the turbine hall, both housing safety relevant equipment necessary to guarantee safety functions needed in case of an earthquake. The basic approach for the evaluation and upgrading of structures at Paks NPP is described in KATONA (2001):

"Design of WWER-440/V213 type twin units has several features, which determine the as built seismic resistance of the units. The box-like reactor building made of reinforced concrete (the containment) was designed for an overpressure of 0.15 MPa, so this building bears the loads caused by a design earthquake. The steel frame turbine hall of 39 m span is connected to the longitudinal gallery building, which is attached to the rigid reinforced concrete part of the confinement. The beams supporting the floors of the gallery building are connected to the wall of the reactor building and the pillars of the turbine hall. The reactor hall, turbine hall and gallery buildings are covered with concrete roof slabs. Seismic resistance of the brick walls separating the different rooms of the gallery buildings is inadequate.

The main building is a set of coupled structures having a separate foundation and widely varying rigidity, and the distribution of the stiffness and masses is highly complex. The problem of optimal modelling of coupled structures with very different characteristics and also the adequate modelling of twin main buildings on a common base mat had to be solved (…)
In the case of the main building structure the soil–structure interaction is modelled through the introduction of the frequency dependent dynamic stiffness matrix obtained for all points of the structural model in contact with the soil, and the equations of motion are solved in the frequency domain. This approach leads to an essential reduction in conservatism compared with the routine calculation methods (...) It seemed to be the most beneficial to stabilize the longitudinal gallery by reinforcement of steel framework of the reactor hall and turbine hall. The idea is to transfer the transversal load from the turbine hall, intermediate building (transverse gallery) and reactor hall to the very rigid reinforced concrete localization towers. This means reinforcement of the roof bend in order to acquire a disk-behavior, reinforcement of the cross braces of the columnes, and transfer of the transversal forces to the localization tower and to a bridge construction connecting them. Thanks to this in the gallery building, which is overfilled by equipment, there is no need to implement modifications and reinforcements. This reinforcement concept allowed considering also the systems in the turbine hall for cooling and long term heat removal as it was mentioned above. The structural fixes of the turbine and reactor halls excluded also the falling-in of the concrete roof panels. Solutions for the increasing of seismic resistance of the structures essentially mean use of new structural elements (e.g. cross braces, reinforcement of the joints) and reinforcement of the main load bearing elements. The implementation of the structural fixes is going on. The total weight of the reinforcement to be added is more than 1,700 t."

Among others, the following buildings and building structures were reinforced (HAEA 2011): Main buildings of units 1-4, Turbine hall podiums of units 1-4, Brick walls of units 1-4 near electrical equipment, main control room suspended ceilings, emergency diesel engine buildings, auxiliary buildings, building of water intake control, control building of demineralised water.

The implemented bridge structure between the bubble towers for reinforcing the reactor building structures is shown in figure[1] 3-1.

As the spent fuel pool is part of the massive concrete block of the reactor building, an additional seismic reinforcement was not necessary (HAEA 2011). Nevertheless, fuel element integrity could be endangered by damage of the roof structure and a subsequent falling down of debris into the pool. Therefore, the seismic protection of the reactor hall was assessed. Reinforcements were implemented to assure the integrity of the reactor hall and to avoid falling down of roof panels.

Also, the stability of the parking position of the refuelling and hoisting machines above the open spent fuel pool was considered. It was concluded that the fall of these machines didn´t need to be assumed. The occurrence of an earthquake concurrent with the displacement of the refuelling and hoisting machines was not assumed due low combined probability. The contribution of such cases to the overall risk was evaluated in the probabilistic safety assessments (HAEA 2011).

Reinforcements of structures can influence their seismic responses including floor response spectra. After developing the structural fixes, the dynamic calculations were repeated for the modified configurations (KATONA & BAREITH 2011). One reason was to check the adequacy of the upgrades and acceptability of upgraded structures in relation to deterministic code requirements. Another reason was the necessity to determine the specific floor response spectra for the evaluation of mechanical components and piping including their anchorages and for the qualification of active systems and equipment (KATONA & BAREITH 2011). This was less important for the stiff reinforced concrete containment part of the main building complex, but it was important e.g. in the gallery buildings where the stiffness could be influenced by structural upgrades. An iterative approach was chosen for the upgrades of buildings containing an evaluation of the influence of certain modifications on the response and resistance of structures. This procedure was also applied to upgrades of the reactor coolant system where fixed configuration has been re-calculated for the justification of code compliance of the integrity. According to KATONA & BAREITH (2011) the Seismic PSA (SPSA) – containing independent walk-downs and analyses – provided a final evaluation of the effectiveness of upgrading measures in terms of CDF values.

It is also worth mentioning that within the framework of the IAEA co-ordinated “Benchmark Study for the seismic analysis and testing of WWER-type NPP's”, dynamic structural testing activities have been performed at Paks NPP (GÚRPINAR & ZOLA 2001). The objective of the experimental investigation was to obtain data on the dynamic behaviour of the plant's major constructions to support the analytical assessment of their actual seismic safety. To generate the experimental data, the site was subjected to low level ground shaking induced

[1] All figures and tables are presented in Annex 1.

by underground explosions. The dynamic responses of the important structures of unit 1 were measured and digitally recorded, with the whole nuclear power plant under normal operating conditions. The free field response was measured concurrently and site-specific geophysical and seismological data were simultaneously recorded too. The instrumentation lay-out in the plant covered the reactor containment building, the above-located reactor hall and one of the nearby coupled chimneys. The results obtained from the experiments were compared to analytical studies performed by different organisations/companies. According to GÚRPINAR & ZOLA (2001) the amplitudes of the calculated response spectra were higher than those obtained experimentally.

For dynamic analyses of the primary circuit, an integrated model was used according to KATONA (2001). It included the reinforced concrete structure of the reactor building together with the primary loops and equipment. Its purpose was to qualify the primary system for a less conservative seismic excitation, and to receive relative displacements of the primary circuit in order to evaluate the possible impacts. For the primary circuit an upgrade by viscous dampers was developed and implemented. Fixing of other piping systems were performed by additional supports and also viscous dampers. The installation of viscous dampers for the primary circuit ensures the avoidance of primary circuit ruptures due to a design basis earthquake (DBE). Beyond the assessment and the qualification of the primary reactor cooling circuit components for the DBE, also the emergency core cooling systems and the active pressure relief system of the hermetic compartments were qualified for earthquakes. Consequently, if, despite the reinforcements, a loss of coolant accident would take place after an earthquake, the main safety functions could be maintained (HAEA 2011).

Re-evaluations showed that the insufficient seismic capacity of essential components was mainly due to their anchorage (KATONA 2001). The analyses indicated that the distances between the pipeline supports were too large, so additional supports and viscous dampers had to be installed. According to KATONA (2001) empirical methods were used for the evaluation of functionality of active equipment (pumps, motors, valves, breakers, etc.). In this context, a main conclusion of the easy-fix project was, that anchorage of the I&C racks and cabinets, accumulators was not adequate (KATONA 2001):

> *"Practically all the safety related electrical and I&C cabinets had to be reinforced with new anchorage at the bottom or with cross brace at the top. Distances of the cable tray supports were too large in all cases and additional supports had to be installed. Due to the weak anchorage, additional anchorage had to be used at some mechanical equipment. The easy-fix project was completed by 1995. The easy-fixes including reinforcement of the brick walls concerned more than 5,500 elements on the four units and were accompanied by building in steel framework of about 450 t."*

While the easy-fix reinforcements mainly aimed at the structural safety and stability of racks and cabinets and at the protection from falling down of brick walls the qualification of the functionality of electrical and I&C equipment was done by empirical methods. For items which could not be qualified empirically, qualification specifications were prepared in the frame of the PHARE project. In total, specifications for 665 elements were compiled (12 rack specifications, 647 relay and 4 tank qualification specifications). In the course of the PHARE project a systematic examination of the behaviour of relays during seismic excitation and the respective consequences was performed (KATONA 2001).

Upgrades also refer to measures that can avoid or limit the secondary effects of an earthquake, e.g. fires, flooding and other interactions (e.g. implementation of the emergency discharge of generator hydrogen and shaft sealing oil, reinforcement of the fire extinguishing systems and systems containing fire hazardous materials) (HAEA 2011). Also certain parts of the fire water system of the NPP were reinforced to establish their independence of the external loops (not designed against earthquakes) and to provide the water supply for the internal circuit of the fire protection system. The Diesel-engine-driven fire water pumps were designed for seismic loads.

As has been explained above, in Paks NPP a lot of additional anchoring and fixing was necessary. According to KATONA & BAREITH (2011) thousands of anchor bolts of different type are installed at the plant. They are grouped into structural anchor bolts and non-structural anchor bolts for fixing of components and distribution systems. The anchor bolts of both classes can be divided further into "as built" and newly installed for the seismic fixes (KATONA & BAREITH 2011).

Concerning the "as built" anchor bolts (structural and non-structural) it is pointed out in KATONA & BAREITH (2011) that they are cast-in-place anchor bolts or groups of bolts with e.g. anchor plates or hook-bolts. For non-structural applications also some grouted anchors are part of the "as built" anchor bolts. Concerning newly installed structural anchorages and the heavy-component anchorages mostly once through anchor bolts with large counter-plates were implemented. In some cases also grouted bolts were applied. With respect to the large number of new or additional non-structural anchorages several types of bolts qualified for seismic loads were selected. They were qualified for the relevant loads and are applicable for cracked concrete. Further details about the evaluation of anchorage behaviour and applied codes and standards are provided in KATONA & BAREITH (2011).

Summarised, a lot of analytical work has been done at Paks NPP to reconstitute the seismic design basis and to evaluate the resistance of the plant against seismic loads. A combination of different methods has been applied to this process. As a result additional hardware has been installed in the plant and seismically "weak" components have been replaced by stronger ones. The seismic upgrades are summarised in table[2] 3-1. Different examples of seismic upgrades are shown in figures[3] 3-2 to 3-4.

Concept for safe shutdown and heat removal

The development of an integral concept for safe shutdown, cool down and long term heat removal for Paks NPP had two main purposes (KATONA & BAREITH 2001):

- to identify structures, systems and equipment necessary for safe shutdown, cool-down and long term heat removal as well as for monitoring the plant status;

---

[2] All figures and tables are presented in Annex 1.

[3] All figures and tables are presented in Annex 1.

- to determine operator actions and outline operational as well as administrative procedures and provisions required to achieve and maintain safe shutdown conditions following a design basis earthquake.

Two different options for seismic upgrade were originally considered (KATONA 2001). The chosen solution demanded the enhancement of the seismic capacity of the longitudinal gallery building that is housing many systems and equipment vital for safety. It was decided that fixing the longitudinal gallery building could be done best by reinforcing the steel frames of the turbine hall and the reactor hall. The option chosen additionally makes use of systems and equipment located in the turbine hall (KATONA 2001). Therefore, the turbine hall had to be fixed too. After additional fixings also the systems placed in the turbine hall could be considered as available for the heat removal after an earthquake.

According to HAEA (2011) all systems, structures and components (SSCs) of the Paks NPP, the structural integrity or operability of which is required for ensuring the basic safety functions during and after an earthquake, have been identified and classified into seismic Safety Classes. Also, those non safety SSCs have been identified, which might jeopardise the fulfilment of a safety function by their failure. The seismic classification of SSCs has been done as follows for Paks NPP (KATONA & BAREITH 2011):

- Seismic Class 1: active systems and components;
- Seismic Class 2: passive structures and components needed for ensuring the basic safety functions during and after a DBE;
- Seismic Class 3: SSCs the failure of which may inhibit the safety functions (interacting structures and components, falling-on, causing fire or flooding, etc.);
- Seismic Class 4: no safety functions and no interaction.

The scope of the seismic safety evaluation and upgrades was set by the regulation (regulatory decrees at that time) as for a new design, covering not only the seismic safety classified SSCs (including interacting items), but the whole scope of safety classified SSCs. This is contrary to the relative alleviations concerning the selection of the SSCs in the context of other re-qualification methodologies. According to KATONA (2001) and KATONA & BAREITH (2011), also the process and system requirements differed from those applied in the margin type assessments:

- cooling after the design base earthquake (DBE) shall be ensured unlimited in time, while a 72 hours requirement is set for usual margin-type assessment;
- it is required to ensure the level of redundancy corresponding to the design philosophy of the plant (3 x 100%) with conformance with single failure criteria instead of considering one success path (as in Safety Margin Analysis – see (IAEA 2009)) and a backup only.

As soon as acceleration beyond a specific set-point is measured at the base mat, the non-upgraded part of certain systems will be automatically separated from the upgraded one by quick-closing valves (KATONA & BAREITH 2011). These systems´ parts do not have function during and after an earthquake (the reason of this measure was to optimise the effort for seismic fixes). At the same time, there is a signal for the control room.

As a result, the reactor shutdown, cool-down, and long term cooling are performed with the same original operational and safety system and essentially in the same way as during other normal or emergency shutdown situations (HAEA 2011):

> *"Reactivity control is performed by the safety and control system and by injecting boric acid to the space above the core by the high pressure emergency core cooling pumps through the venting of the reactor head.*
>
> *Initially, the removal of decay heat is performed with the secondary steam blow-down system into the atmosphere or by the opening of safety valves of the steam generators and injection of demineralised water, if necessary. Later, in the lower temperature range, the normal operational cool-down system removes the residual heat. Each component of the cooling technology was qualified and reinforced if necessary.*
>
> *Cooling down and borating should be performed during natural circulation of the primary coolant. Systems required for this function were qualified for the design basis earthquake and reinforced if necessary."*

As pointed out in HAEA (2011) it was demonstrated via shaking table testing of the safety and control assemblies that they would not lose their functionality even if a load significantly higher than that induced by the DBE would occur. Therefore shutdown is not endangered in case of an earthquake.

Among others also the safety power supply and the essential service water system (necessary also for the operation of the emergency diesel generators) were re-qualified and reinforced (HAEA 2011). Also those elements of the spent fuel and refuelling pools that are necessary for continuous circulation of the coolant or to avoid loss of coolant, were analysed. In case it was found necessary they were re-qualified and reinforced (HAEA 2011).

Instrumentation and operator actions in case of an earthquake

According to KATONA & BAREITH (2011), there is no direct initiation of an automatic shutdown of the reactor in case of an earthquake (e.g. by exceeding certain accelerations measured in the plant) in Paks NPP. This decision was based on the analysis of the frequency of expected events, probability and consequences of spurious signals, and the international practice. According to HAEA (2011), an unjustified/spurious shutdown and disconnection of all of the four units from the electric power grid at the same time may have more severe safety consequences than a somewhat delayed shutdown of the reactors due to any abnormal technological signal or due to damage caused by the earthquake.

In case of an earthquake two scenarios are possible in principle:

- The earthquake results in a transient (e.g. due to a failure of certain equipment or spurious I&C signals) and, consequently, in an automatic shutdown initiated by the reactor protection system (RPS). Immediate actions to control the plant and to ensure the necessary safety functions are actuated by the RPS. The subsequent plant status depends on the amount of damage and the sequence of actuations by the RPS. The necessary operator actions depend on the plant status. They are defined by emergency procedures and trained on the simulator (KATONA & BAREITH 2011). It is assumed that operator actions start after the first sequence of automatic actions, i.e. no grace time is assumed.[4] In any case operator actions are needed in the medium term for the implementation of the procedures for cool-down and continuous cooling of the reactor. According to HAEA (2011) the number of operating personnel was determined to be able to carry out the interventions in response to the earthquake (design basis earthquake affecting all four units).

- The plant remains in operation. In case of an earthquake not leading to automatic shutdown the operators have to check whether the criteria for the operating basis earthquake (OBE) have been exceeded. In case of exceedance the plant has to be shut down and to be inspected with respect to possible damages.

The characterisation of the OBE is based on the concept of the “cumulative absolute velocity” (CAV)[5] in combination with response spectrum criteria (KATONA & BAREITH 2011). The OBE-exceedance criteria is set CAV=0.16 gs and response spectrum in the amplified range less than 0.2 g. In KATONA & BAREITH (2011) this criterion for OBE exceedance is judged to be very conservative as the allowed earthquake excitation level is with safety factor three lower than the damage limit of ordinary structures not designed for earthquake. According to EPRI (2006) the value of CAV=0.16 gs was found to be a conservative characterisation of the threshold between damaging earthquake motions and non-damaging earthquake motions for buildings of good design and construction as defined by the Modified Mercalli Scale.

As pointed out in the National Action Plan (NacP) published by HAEA (2012), the question of seismic shutdown should have been re-evaluated in the frame of the reconstruction project of the seismic instrumentation, which is in a preparatory phase. The respective final deadline for this re-assessment was December 31, 2012.

Seismic PSA

A seismic PSA (SPSA) has been compiled for Paks NPP, respective information is contained in BAREITH et al. (2003), BAREITH (2007); ELTER (2010); KATONA & BAREITH (2011) and HAEA (2011). The objectives of the SPSA were to determine the core damage frequency (CDF) due to seismic loads, to identify plant vulnerabilities and to evaluate the consequences of different improvements (in terms of CDF values) (BAREITH et al. 2003). Different versions of the SPSA have been prepared. They account for different states of seismic upgrades. Moreover containment performance was also evaluated to enable extension to a level 2 PSA.

---

[4] According to KATONA & BAREITH (2011) except for the reactor protection actions, a “concept of a full automatic cool-down technology with (let say 30 minutes) grace-time might be questionable in case of a serious earthquake. The earthquake can cause several initiating events (breaks, losses of fluid, losses of power etc.). The variability of combination of simultaneous disturbances/damages, i.e. common cause failures is much more than it could be foreseen and programmed in.”

[5] The CAV is defined as integral of the recorded seismically induced accelerations (above a certain value) over time (EPRI 2006).

The SPSA study originally covered plant operation at full power - unit 3 was selected as a reference for the analysis (BAREITH et al. 2003). The available information concerning the influence of different upgrades refers to operation at full power. In HAEA (2011) results of a SPSA for shutdown modes are provided (in terms of CDF values), however, no further details are presented.

For the compilation of the SPSA, it was necessary to assign exceedance probabilities to a wide spectrum of peak ground accelerations (PGA). Therefore, in addition to the hazard analysis for the design basis earthquake, an additional hazard analysis was performed by a local institute (BAREITH 2007). As is pointed out in BAREITH (2007), the results of the hazard analysis performed in course of the PSA agree well with those of the evaluation of the design basis earthquake (PGA = 0.25 g at around 10,000 years return period) which was the basis for the seismic upgrades of Paks NPP.

Based on the assessment of the soil characteristics of the site of Paks NPP (soft soil) the possibility of occurrence of soil liquefaction cannot be excluded (HAEA 2011). The analyses to determine the seismic hazard also contained investigations concerning the soil liquefaction potential in addition to the evaluation of the spectral accelerations.

The mean frequencies of the seven acceleration ranges that have been used as input for the SPSA are presented in table[6] 3-2. The lower bound of 0.07 g corresponded to the lowest HCLPF capacity (HCLPF: high confidence of low probability of failure)[7] for all structures and equipment at the time of the analysis while the upper bound of 1.0 g is the highest acceleration evaluated in seismic hazard analysis (BAREITH 2007). According to BAREITH (2007) the bounds of the acceleration ranges were set the way that the seismic hazard curves remain approximately linear throughout a range. The intervals are larger for higher accelerations which is due to the fact that the accelerations change more slowly at lower exceedance probabilities.

In addition to the evaluation of the seismic hazard, the SPSA required the determination of the fragilities of the relevant SSCs. Fragilities represent the probabilities that certain buildings or components fail due to a seismic load. Fragility values are usually expressed in terms of multiples of PGA. The failure probability depends on the confidence level (see figure[8] 3-5).

Fragility analysis for the SPSA was done by experts from ABS Consulting (USA) (KATONA & BAREITH 2011). It contained reviews of design criteria and design reports. For the determination of fragilities, the scaling method defined in IAEA (1993) was used. Results of analyses and of tests performed in the course of the seismic (re-)qualification as well as an evaluation of seismic ruggedness on the basis of seismic experience were taken into account. Also, the analyses took into consideration the effect of seismic upgrades of structures and structural elements (KATONA & BAREITH 2011). In most cases the fragilities were obtained by scaling the results of already existing analyses of structures and equipment to the median acceleration capacity A_m. For the determination of the

---

[6] All figures and tables are presented in Annex 1.

[7] The HCLPF value represents the PGA value for which the respective SSC has a failure probability of at most 5% with a confidence level of at least 95%.

[8] All figures and tables are presented in Annex 1.

uncertainty of the median capacity the respective uncertainties of the important variables contributing to the overall fragility were taken into account. All variables were assumed to be log-normally distributed. Further details concerning this approach are provided in KATONA & BAREITH (2011).

Due to the large number of SSCs it would have been overly complex to assign specific fragilities to each SSC. However it also was not necessary for components expected to fail at accelerations high enough so as to the CDF is dominated by components with lower seismic capacity. Therefore, SSCs were screened with respect to their seismic capabilities. Seismic experience based screening was used. It relies on the collection and evaluation of the performance of similar equipment in strong motion earthquakes. SSCs with comparably high capabilities were screened out. For those structures and components that could not be screened out, detailed fragility calculations were needed in the course of the SPSA. An important part of the screening analyses were assessments considering observations during seismic walk-downs (KATONA & BAREITH 2011). Further details concerning the guidelines used for plant walk-down and screening are provided in KATONA & BAREITH (2011).

In the course of the screenings, parts of the mechanical, electrical and I&C components were assigned to three screen categories. These groups contain equipment that is inherently rugged or has been upgraded (BAREITH et al. 2003). No detailed assessment of the seismic capacities of these components was performed (BAREITH 2007):

> *"The analysis of seismic response based on the results of finite element evaluation of structures and floor response spectra were already available from SMA for practically all levels of interest within the buildings of safe shutdown components. Comprehensive walk-downs were conducted to examine all structures and plant locations that contain mechanical equipment, piping, switchgear, electrical equipment, instrumentation and control cabinets, and cables that may affect any systems or functions that are analysed in the PSA. Systematic screening criteria were applied during the walk-downs to determine whether the seismic capacity of an examined item is sufficiently high to justify no need for additional evaluation. As a result of an initial evaluation process, many structures and components were assigned to three screening groups, based on their assessed capacity."*

For the high screen category, common fragilities were assigned to the mechanical and electrical and I&C components. According to BAREITH (2007) this category was defined by the criteria HCLPF = 0.53 and median acceleration capacity A_m = 1.34 g.[9]

For the low screen category different fragility values were assigned to mechanical components on the one hand and to electrical and I&C components on the other hand (BAREITH 2003; 2007). The low screening criteria for mechanical equipment were HCLPF = 0.35 g and A_m = 0.89 g. For tested relays (contact devices) and cabinets the relay screening capacity criteria HCLPF = 0.27 g and A_m = 0.73 g were used (BAREITH 2007).[10]

[9] In KATONA & BAREITH (2011) slightly different values are provided.

[10] In KATONA & BAREITH (2011) slightly different values are provided.

According to BAREITH (2007) the failure fractions for each screening group were determined by their respective fragility distributions. The specific influence of each group was determined by the most limiting combination of failures of the specific structures and equipment included in the group. Concerning the components assigned to the high screen category, their failure is assumed to occur fully correlated in the PSA model, which is a conservative assumption. The results of the Paks SPSA show that the seismic failure of all high screen SSCs provides no important contribution to the seismic risk (KATONA & BAREITH 2011). The reason is that the HCLPF value threshold for the high screen level is about 0.5 g PGA as compared to safe shutdown earthquake of 0.25 g PGA which was the basis for the seismic upgrades.

For structures, mechanical equipment, electrical equipment, cabinets and relays that could not be assigned to one of the above mentioned screening categories specific analyses were performed to determine their capacities. For mechanical components existing deterministic analytical evaluations were examined to determine if a specific fragility had to be developed based on the scaling method described in IAEA (1993) or if a generic value could be assigned based upon the screening of a similar component or the calculation of fragility for a similar component (KATONA & BAREITH 2011). For electrical and control cabinets as well as instrumentation racks the screening of the structural integrity could be performed according to the two screening criteria mentioned above and the observed anchorage. This was not possible for the screening of active contact devices in the cabinets. As these components could not be screened on the basis of seismic experiences results of tests performed under the PH2.04/94 PHARE Program were used for the fragility calculations (KATONA & BAREITH 2011). For untested active contact devices (relays) and for cabinets with low anchorage capacity, a low capacity was assumed and a generic calculation was performed to determine the respective fragility.

Again similar types of equipment could be assigned to certain groups, if similar seismic capacities were determined and they were expected to fail at approximately the same acceleration level (BAREITH 2007):

> *"Over and above the one higher screen and the two lower screen groups, the following separate seismic failure groups were defined – with the associated fragilities – for modelling and quantification in PSA:*
>
> *27 groups of mechanical equipment, grouping based on equipment type and/or location.*
>
> *9 groups of electrical and I&C cabinets, grouping based on cabinet location.*
>
> *20 groups of electrical and I&C relays (contact devices), grouping based on relay type.*
>
> *(…)*
>
> *Fragilities were developed using the standard separation of variables approach and they were mostly based on existing deterministic analyses conducted during the upgrading. The focus was on consequences of liquefaction, non-ductile failure modes of steel structures and spatial systems interactions identified during the walk-down."*

According to BAREITH (2007) 11 different structural failure modes due to seismic loads were examined. An overview of seismic failure modes for the electric power supply and the ultimate heat sink is provided in HAEA (2011) (sections 2.2.1.2 and 2.2.13).

Soil liquefaction could act as an important initiator for common cause failures (CCF) concerning the availability of the electrical power supply and the ultimate heat sink (HAEA 2011). Site-specific analyses were performed to determine the likelihood of soil liquefaction. The potential damage to structures, equipment, buried piping, cables, etc. due to two degrees of liquefaction was evaluated. Evaluation of the behaviour of the safety-classified buildings of the NPP in case of soil liquefaction concluded that it does not lead to a loss of overall stability of the buildings, but may cause the settlement of the buildings (HAEA 2011). An island-like settlement of the main building complex could damage the systems connected from outside to the main building such as underground pipelines and electric cabling (also for emergency diesel generators).

The effects of seismically induced failures were taken into account as transient initiating failures and as mitigating system/component failures. These failures were determined by an evaluation of failure consequences for each component within a group. The consequences of a simultaneous occurrence of different group failures were identified. In this context, a list of transient initiating failures that can potentially occur due to an earthquake was established. For most of them there were similar initiating events in the internal event PSA. But there were also some specific initiating events not modelled in the internal event PSA because of the low likelihood of occurrence due to internal events. In case of an earthquake their likelihood was no longer negligible due to some earthquake specific effects like e.g. relay chatter. According to BAREITH (2007), a comprehensive and detailed analysis of electrical circuit diagrams was performed to determine the consequence of I&C failures including seismic induced chattering of contact devices. The consequences of a collapse of block walls on electrical cables were also taken into consideration.

At the time the baseline SPSA was compiled, test results were not available for all electric components. As is pointed out in ELTER (2010), correlated failures of the untested relays and cabinets were assumed in the baseline SPSA and a single element was used in the model to describe these correlated failures, i.e. the 29 groups of cabinets and relays were assigned one single fragility value and treated as one large group (BAREITH et al. 2003). The failure of this group had a serious CCF potential as it could cause loss of offsite power and failure of the diesel generators, inadvertent closure or opening of all steam generator isolation valves, inadvertent opening of all steam generator safety valves and failure of all feedwater systems as well as failure of all emergency core cooling systems (ELTER 2010; BAREITH 2007).

Based on the exceedance probabilities for different PGA values and the failure probabilities (fragilities) of the relevant SSCs, models for the possible accident sequences had to be developed. The main objective was to construct an event tree structure that integrates event trees developed earlier within the internal initiator PSA study and specific earthquake-induced transients into a generic model that reflects the specifics of an earthquake (BAREITH et al. 2003). The process of model development contained the following major steps (BAREITH et al. 2003):

"-selection (grouping) of equipment level failures that can be caused by different seismic-induced failures (groups)

-identification of transient initiating failures and additional system, train or component level failures and degradations that can be caused by any combination of equipment failures selected for a group, establishment of a list of transient initiating failures that can be caused by an earthquake

-development of functional event trees for single transient initiating failures

-development of a generic event tree for modelling plant responses to an earthquake with combinations of single and multiple transient initiating failures

-modeling containment performance"

One purpose of the SPSA was to quantify the effectiveness of seismic upgrades with respect to CDF values. According to BAREITH et al. (2003), revised versions of the SPSA were based on the assumption that the envisaged upgrades result in an increase of seismic capacities. These assumptions refer to upgrades of structural fixes and an increase in seismic capacity of relays and electrical cabinets (KATONA & BAREITH 2011). According to KATONA & BAREITH (2011), only such upgrades were taken into account, that were judged to be technically feasible and the assumed increased capacity was considered achievable by the experts of ABS Consulting (KATONA & BAREITH 2011):

"The assumed upgrade of structural elements, e.g. reinforcement of bolted connections and masonry block walls, were similar or identical to fixes that had already been made on other similar structural elements in the plant during the Seismic Safety Program. Also, for relays and electrical cabinets only such upgrades were assumed, e.g. improvement of anchorage capacity, seismic qualification and testing of relays (or replacement at worst), that had previously been applied to thousands of similar components within the Seismic Safety Program. In effect, most of these upgrades have since been implemented at the plant."

In UMWELTBUNDESAMT (2006) information on the state of the upgrades and quantitative information on the reduction achieved by upgrades up to that time and to be achieved in the future was mentioned as desirable. Information concerning these aspects is contained in ELTER (2010) and BAREITH (2007).

In ELTER (2010) the main motivations for different upgrades and their respective consequences with respect to seismic safety are discussed. According to ELTER (2010) the results of the baseline PSA were strongly influenced by structural failures of bolted connections in the turbine building leading to eventual collapse of the building. Subsequent steam and feedwater header ruptures were assumed. A total loss of main and auxiliary feedwater disabling heat removal through the secondary side via the closed steam and feedwater circuit would be the consequence. The emergency feedwater may remain available allowing for an open circuit heat removal with release of the steam generated in the steam generator to the environment. Continuous make-up of the secondary circuit water volume is necessary in this case. This option was assumed to be insufficient to ensure long term heat removal. Therefore, the ultimate measure for heat removal from the core in such a situation was primary feed-and-bleed with water supply to the emergency core cooling system heat exchangers via the service water system. It was decided that the most effective upgrade was to increase the structural capacity of the Turbine Building bolted connections.

In BAREITH (2007) quantitative effects of the different steps of upgrades are presented:

- Upgrade 1 increases the structural capacity of the turbine building bolted connections and the reactor hall/longitudinal electrical gallery bolted connections. The revised PSA is based on the assumption that the seismic capacities of these connections are increased to the extent that failures of these structures are now governed by buckling of the vertical frames. The effects from this upgrade reduce the total core damage frequency by a factor of approximately 3, compared with the baseline results.
- Upgrade 2 increases the structural capacity of all interior masonry block walls that are located near PSA equipment and cables. Correlated failures of these walls are modelled in the baseline PSA by a separate structural failure event. The revised PSA is based on the assumption that the seismic capacities of all walls in this group are increased to at least the lower screening capacity for other plant structures and mechanical equipment (i.e., HCLPF = 0.35 g, median = 0.89 g). The effect from this upgrade reduces the total core damage frequency by approximately 4%, compared with the baseline results.
- Upgrade 3 increases the capacities for all untested relays and cabinets that affect any equipment in the PSA models. The revised PSA is based on the assumption that the seismic capacities of all kinds of cabinet anchorage are increased to at least the lower screening capacity for other plant structures and mechanical equipment (i.e., HCLPF = 0.35 g, median = 0.89 g). The revised analysis is also based on the assumption that all relays are either tested to demonstrate that their capacities meet or exceed the minimum screening capacity for all other tested relays (i.e., HCLPF = 0.27 g, median = 0.73 g), or the relays are replaced with qualified components that exceed this capacity. The combined effects from upgrades 2 and 3 reduce the total core damage frequency by approximately 9%, compared with the baseline results.
- Upgrade 4 increases the structural capacity of the diesel generator building 14 cm interior block walls. The revised PSA is based on the assumption that the seismic capacity of these walls is increased to at least the lower screening capacity for other plant structures and mechanical equipment (i.e., HCLPF = 0.35 g, median = 0.89 g). The combined effects from upgrades 1, 2, 3 and 4 reduce the total core damage frequency by a factor of 5.6, compared with the baseline results.
- Upgrade 5 increases the structural capacity of the turbine building vertical braced frame. The revised PSA is based on the assumption that the combined seismic capacities of the turbine building frame and the bolted connections are increased to at least the lower screening capacity for other plant structures and mechanical equipment (i.e., HCLPF = 0.35 g, median = 0.89 g). The combined effects from upgrades 1, 2, 3, 4 and 5 reduce the total core damage frequency by a factor of approximately 8, compared with the baseline results.

- Upgrade 6 increases the structural capacity of the diesel generator building 30 cm block walls. The revised PSA is based on the assumption that the seismic capacity of these walls is increased to at least the lower screening capacity for other plant structures and mechanical equipment (i.e., HCLPF = 0.35 g, median = 0.89 g). The effect of this upgrade is evaluated in combination with upgrades 1, 2, 3, 4, 5 and 7 below.
- Upgrade 7 increases the structural capacity of the air compressor building[11]. The revised PSA is based on the assumption that the seismic capacity of this building is increased to at least the lower screening capacity for other plant structures and mechanical equipment (i.e., HCLPF = 0.35 g, median = 0.89 g). The combined effects from upgrades 1, 2, 3, 4, 5, 6 and 7 reduce the total core damage frequency by a factor of 10, compared with the baseline results.

While BAREITH (2007) provides a comprehensive overview of the quantitative aspects of different upgrades, the combined effect of the upgrades 1 to 7 which is mentioned there (factor 10) does not correspond to actual SPSA results. According to ELTER (2010) the CDF contribution due to seismic events was around 2.9×10^{-4}/a in the baseline PSA (see figure[12] 3-6). According to HAEA (2011) the actual value is 4.31×10^{-5}/a for power operation which corresponds to a reduction by a factor 6.7.

In HAEA (2011), an actual distribution of CDF values over the range of accelerations for power operation is also provided (see table[13] 3-3).

According to HAEA (2011), the expected value of the annual core damage frequency from earthquakes occurring in a shutdown state is 4.72×10^{-6}/year, considering each shutdown service state and acceleration range.

According to KATONA & BAREITH (2011) calculations taking into account the change of failure probabilities of SSCs with the level of confidence were performed. This was done by sampling of failure fractions on the basis of convoluting randomly selected hazard and fragility curves from a representative set of confidence levels over the seismic accelerations. To do so, a representative range of confidence levels of the fragilities was used. The resulting probability distribution of core damage frequency (CDF) shows the effect of the different confidence levels and thus the overall confidence in the estimated risk level.

Remaining safety relevant aspects according to the EU Stress Tests

The following compilation of remaining safety aspects is based on the information given in the National Report of Hungary on the Targeted Safety Reassessment of Paks Nuclear Power Plant (HAEA 2011) with additional information from the National Action Plan (NAcP) (HAEA 2012).

The possibility of occurrence of soil liquefaction cannot be excluded based on the assessment of the soil characteristics of the site (HAEA 2011). Simple empiric or semi-empiric methods (leading to conservative results) showed a safety margin against soil liquefaction for pore water pressure causing the liquefaction

---

[11] Damage of the compressor building of high pressure instrumentation air system leads to loss of high pressure instrumentation air, and thereby to the spurious close of steam generator isolation valves after some time (HAEA 2011).

[12] All figures and tables are presented in Annex 1.

[13] All figures and tables are presented in Annex 1.

in the layers between 10 and 20 m of only approximately 1.1 (HAEA 2011). Stress calculation with more detailed methods utilised in the most recent studies resulted in larger margins, since non-linear effects are more effective in the range of stronger earthquakes. According to the results of calculations considering the soil-building interaction that modifies the cyclic shearing stress, the recurrence period of seismic events leading to soil liquefaction in the layers loaded by the main building is 14,000-18,000 years. According to the current Hungarian regulation the DBE has a recurrence period of 10,000 years. Therefore soil liquefaction has not to be considered in the context of the design basis. However as the dominant contribution to structural failures for accelerations beyond the design basis value is due to soil liquefaction additional measures are necessary according to HAEA (2011). Settlement of building could damage underground lines and connections. Therefore they have to be re-qualified or, if necessary, to be modified to allow for relative displacements. Investigations are currently going on to more accurately assess the potential building settlement after an earthquake.

It is pointed out in HAEA (2011) that in the lower acceleration ranges soil liquefaction causing settlement of the main building complex plays a dominant role in the occurrence probability of total loss of electric power supply and loss of ultimate heat sink. In the case of earthquakes up to a PGA of ≤ 0.30 g (slightly above the design basis of 0.25 g) the mean probability for the occurrence of a total loss of electric power supply is 9%, the mean probability for the occurrence of a loss of ultimate heat sink is 11%. If the capacity of the relevant structures to withstand soil liquefaction could be enhanced by reinforcements to the level of the tested relays and cabinets (HCLPF = 0.27 g), the mean probability for the occurrence of a total loss of electric power supply would be reduced to 5%, the mean probability for the occurrence of a loss of ultimate heat sink to 6%.

According to the NAcP (HAEA 2012), the underground lines and connections (pipelines, cables) at risk due to potential settlement of the main building shall be re-qualified and, if necessary, modified to allow for a relative displacement. The final deadline is December 15, 2017. Additionally, a state-of-the-art analysis for the assessment of the existing margins of earthquake-initiated building settlement and soil liquefaction phenomenon shall be performed (final deadline December 15, 2018).

Aside from the issue of soil liquefaction, during a review of the design basis some aspects became apparent, which could influence the fulfilment of main safety functions or appropriate actions by the staff after an earthquake (HAEA 2011). According to the Hungarian Atomic Energy Authority (HAEA) they either should be further investigated to evaluate their safety significance or measures should be taken to increase the level of safety:

- A damage to the service building, which is a reinforced concrete structure building of significant robustness, but not formally qualified against DBE loads, could lead to failure of the three common demineralised water storage tanks of Installation II (e.g. units 3 and 4). The tanks are situated in the direct vicinity of the service building. They can withstand the direct seismic loads but they are not protected against the falling down of panels from the service building. As a consequence, the possibility that one of the tanks (that back up each other) could fail in case of an earthquake was taken into account in the seismic PSA. Additionally, the personal dosimeters are stored in the service building and the change to protective cloths will also be more difficult in case of a failure. The usability of the building therefore also has logistic importance. According to the NAcP the walls of the building shall be seismically qualified and, if necessary, reinforced or provide appropriate protection of the tanks by other means (final deadline December 15, 2015).

- The building for the plant fire brigade is not qualified against earthquake loads. As the barrack building is made of reinforced concrete, the protection of the personnel and the equipment in this building could be ensured with minor measures. According to the NAcP, some intervention is necessary to protect the personnel and equipment in the fire brigade headquarters (final deadline December 15, 2015).
- The components of the main condenser cooling water system including steel pipelines of 3600 mm in diameter are placed in a trench. The system is not relevant for heat removal in case of a DBE and is not qualified against earthquake loads. Therefore, potential damage of the pipelines cannot be excluded even for loads less than the DBE. In case of a failure, the water contained in the pipelines can theoretically be accommodated by filling up the trenches. However, the extent of flooding and other effects need further investigations in case of the main condenser cooling water pumps not being stopped by chance. In this case, the uniform filling up of the trenches is not guaranteed and local flooding might occur. Depending on the location of damage, flooding of the safety cable tunnels towards the emergency diesel generator and water intake buildings, the cellar level of the water intake building containing safety cable junctions and of the cellar level of the turbine hall might be also possible. The consequences from the rupture of large diameter pipelines should be investigated and the protection against such an event should be improved, if necessary. As a measure envisaged to increase the robustness of the plants against earthquakes, installations to stop the main condenser coolant pumps when the main condenser coolant pipeline is damaged are mentioned in HAEA (2011). It should be ensured that the pipeline trenches are able to receive and drain the discharged water. If necessary, the slope has to be elevated or a protective dam has to be constructed to avoid the flooding of the turbine hall or the cable tunnels. According to the NAcP the final deadline for these measures is December 15, 2015.
- Concerning the function of the essential service water system and therefore the availability of the ultimate heat sink and cooling main emergency diesel generators, it was recognised that seismic qualification of the filtration units (machine racks and travelling water band screens) for screening the centimetre and millimetre large pieces at the essential service water pumps was not part of the former scope. Independent of the robust structure of the filtration units, further investigation concerning their performance after seismic events is needed. Possibly additional measures have to be implemented to avoid clogging after an earthquake. According to the NAcP, it shall be analysed if the lack of seismic qualification of the machine racks and travelling water band screens of the essential service water system jeopardises the ultimate heat sink function and, if necessary, the adequate exclusion measures shall be implemented (final deadline December 15, 2015).
- Further improvements envisaged to increase robustness of the plants against earthquakes concern the external power supply via the 400 kV and 120 kV grids, the seismically safe storage of maintenance tools and equipment stored at the units after the outages, seismic qualification of shelters for staff, internal and external communication. According to the NAcP, the protection

of the not seismically reinforced 400 kV and 120 kV substations and the automatisms switching the plant to isolated operation against earthquakes shall be evaluated and increased if necessary (final deadline December 15, 2014).

- As already mentioned, a LOCA is not expected due to a DBE as the primary system has been re-qualified for the respective seismic loads. However, in case the occurrence of a seismically induced LOCA is assumed, the available emergency operating procedures for LOCA are judged to be not optimal, but still effective. As further improvement envisaged, the available symptom-based emergency operating procedures have to be reviewed so that they support the optimal recovery after simultaneous occurrence of an earthquake and rupture of the primary coolant circuit. According to the NAcP the final deadline for this re-assessment is December 15, 2013.
- The database containing seismic safety classification of the components has to be reviewed to provide that the classification is in agreement with the information given in the licensing documentation of seismic safety improvement modifications. According to the NAcP the revised database was completed by 30^{th} April 2012 and its regulatory supervision was also performed.
- According to the NAcP, the seismic-proof fixing of temporary, non-process equipment in the outage and recovery of fixings dismantled for maintenance purposes are not duly regulated. Therefore extraordinary attention shall be paid to seismic-safety related housekeeping and full recovery of fixings after main outages. Fixing of the non-process equipment and maintenance tools that could adversely impact process equipment during outages shall be provided (final deadline December 15, 2014).

An additional aspect mentioned in UMWELTBUNDESAMT (2006) concerned the possibility of a flood caused by a postulated 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 was judged to be of interest, including an analysis of possible consequences for the nuclear power plant, and, if applicable, of counter-measures. This aspect is addressed in chapter 3.1.1 in HAEA (2011). According to HAEA (2011), several conservative assumptions have been combined for the analysis of the consequences of a postulated break of dam of the hydro power plant of Gabcikovo. Also, this assessment covers the case when the dam breaks due to an earthquake. Since the highest water level at the section of Paks NPP is by 1 m less than the level of the site, flooding hazard needs not be taken into consideration (as additional conservatism it was not taken into account that the level on the left bank is 0.34 m less than the maximum water level (see also figure 3-1 in HAEA (2011)).

In the peer review country report on the Stress Tests performed for Paks NPP the reviewers acknowledged the measures undertaken to upgrade the plant to its current standard (ENSREG 2012). The combination of repeated seismic hazard analyses and subsequent retrofitting measures were judged to be among the “best practices” identified during the Stress Tests.

The reviewers concluded that the robustness of the plant against earthquakes has been significantly increased by the implementation of the seismic safety upgrade programme. It is mentioned that additional safety upgrading measures are envisaged. The reviewers recommended to the Regulator to monitor the implementation of the measures for strengthening of the level of protection of the plant structures against liquefaction effects and soil settlement, as well as for the completion of seismic qualification of certain SSCs and a review of the database containing the seismic safety classification of components.

3.2 Evaluation and conclusions

Information obtained from the Hungarian side is sufficient to trace the process of seismic hazard assessment, hazard updates and reviews. The topic has been clarified. The topic “Seismic hazard assessments for siting and licensing of Paks 5&6” has been clarified by the Hungarian response.

The information obtained on assessment of Quaternary faults in the near-region of the site is not yet satisfactorily clear.

It is evident that previous seismic hazard assessments included serious efforts to identify Quaternary faults in the vicinity and near-region of the site. Evidence of Quaternary surface-breaking faults in the vicinity and near-region of the site has been obtained from high-resolution reflection profiling in the early phase of seismic hazard assessment (MAROSI 1997; TÓTH 2003). The general existence of active faults in the region has recently been confirmed by other studies (MAGYARI 2011). However, documents and materials supplied are not sufficiently clear in describing whether these faults have been further studied or not. Documents do not specify whether or not an adequate effort was undertaken to constrain the youngest fault slip history and assess prehistoric earthquakes, which may have occurred at these faults.

The topic appears particularly important with respect to the location of the plant in the vicinity of faults paralleling the Mid Hungarian Fault Zone, which has been identified as an active seismogenic source by investigations of Hungarian geoscientists (LÖRINCZ et al. 2002; HORVATH 2004; BUS 2009). At this background, a systematic assessment of Quaternary faults and the parameterisation of slip history, youngest slip events, fault geometry and slip velocity is of utmost importance for the reliability of seismic hazard assessments.

The Austrian experts therefore renew and extend their information request regarding seismic hazard assessment.

Originally, Paks NPP was not designed against seismic loads. A large effort was undertaken to upgrade the plant to the level of the DBE defined in the course of an updated seismic hazard assessment (PGA = 0.25 g). This was a complex task. The selected approach and the different upgrades are widely described in the referenced documents. They also contain extensive information on basic model assumptions applied to the Seismic PSA. Therefore, sufficient information is available for the kind of plausibility check to be performed in the course of the Roadmap.

Our evaluation focussed on the methodologies applied to seismic upgrades (mainly: application of design codes like ASME or KTA vs. seismic margin assessment) and on the definition of the screening criteria for equipment applied in the context of the SPSA. Our respective questions compiled after the 16th bilateral meeting in 2010 have been answered by Hungarian experts in a comprehensive manner, so it is has been possible to gain a sufficient perspective on these issues.

It is evident that it was not possible to demonstrate full scope code compliance for Paks NPP in the course of the seismic re-qualification. The reason is that the already built plant had not been qualified for seismic loads during design and construction. Therefore, the selected “mixed” approach, using procedures and criteria usually applied for a new design in combination with methods and techniques developed for seismic re-evaluation of operating nuclear power plants, is plausible. According to the information given by the Hungarian experts, the effectiveness of upgrades was evaluated. The consequences of structural upgrades with respect to the dynamic answers of structures (floor response spectra) were also assessed. Consequently, no open questions remain concerning the methodologies applied to seismic upgrades. The same is true for the criteria used to screen out mechanical and electrical components in the context of the compilation of the SPSA.

In the process of seismic upgrading the effects of specific measures to be implemented were quantified by different versions of SPSA taking into account these measures. SPSA was systematically used as an important analytical tool, which is a reasonable approach in the course of the implementation of seismic upgrades in Paks NPP. According to the information given in the referenced documents also specific failure modes, e.g. spurious signals due to I&C components which were not seismically qualified, were taken into account.

Regularly the required robustness against seismic loads should be implemented in the design stage for of a predefined DBE on the basis of compliance with the relevant codes and standards. This usually implies that some margins exist to accommodate higher loads than those induced by this DBE. The reason is some inherent conservatism of the applied methods (e.g. mainly elastic analysis on the basis of prescribed stress values instead of allowance for energy dissipation due to plastic deformation). The respective reserves can be quantified with the aid of SMA methods (whereby one has to be aware that engineering judgement plays a much larger role in the course of SMA than for design against standard codes).

The existence of an inherent reserve against loads beyond the actual design basis does not apply to Paks NPP. As the plant was not designed against seismic loads, reserves already had to be utilised for re-establishing the design basis. According to the results on the SPSA for Paks NPP, there is only a low conditional probability for a core damage for seismic loads up to the updated design basis (PGA = 0.25 g). This reflects the effect of the seismic upgrades. On the other hand, there are no significant reserves for PGAs above this value which is illustrated by the following results of the SPSA:

- For earthquakes up to a PGA of ≤ 0.30 g (slightly above the design basis of 0.25 g) the mean probability for the occurrence of a total loss of electric power supply is actually 9%, the mean probability for the occurrence of a loss of ultimate heat sink is 11% (HAEA 2011).

- The conditional probability for core damage is 8.7% for the acceleration range from 0.22 g to 0.32 g while it is 97% for the acceleration range from 0.32 g to 0.48 g (see table[14] 3-2).

The results presented in the Environmental Impact Study (EIS) in 2006 showed that seismic events dominated the overall core damage frequency at that time with a value of 2.6 x 10^{-4}/year and unit (EIS 2006). In the report to the Austrian Government on Paks NPP lifetime extension (UMWELTBUNDESAMT 2006) it was judged that this value was significantly higher than the target value for CDF for existing nuclear power plants as it has been formulated by the International Nuclear Safety Advisory Group of the IAEA. According to up-to-date information contained in HAEA (2011), the expected value of the annual core damage frequency from earthquakes occurring during power operation is 4.31 x 10^{-5}/year, while for shutdown states it is 4.72 x 10^{-6}/ year. Therefore, the total CDF value due to seismic events is 4.8 x 10^{-5}/year which is by a factor of 5.4 lower than the value presented in the Environmental Impact study (EIS 2006). This reflects the effect of the seismic upgrades. According to PSA results presented in ELTER (2010), the actual overall CDF value including earthquakes is below 1.0 x 10^{-4}/year. The development of the relative importance of different initiators as well as their absolute numbers can be seen in figure[15] 3-6. Apparently, seismic events still provide the dominant contribution to the CDF.

According to KATONA & BAREITH (2011), no direct automatic shutdown is initiated in Paks NPP in case of an earthquake (e.g. due to the exceedance of certain accelerations). As far as the earthquake results in a transient or damage to the plant automatic shutdown is initiated by the reactor protection system due to signals generated by these initiators. This approach corresponds to practices adopted also in other countries. As pointed out in the NAcP (HAEA 2012) the question of seismic shutdown had to be re-evaluated in the frame of the reconstruction project of the seismic instrumentation until December 31, 2012. Results of this re-evaluation would be of interest.

In KATONA & BAREITH (2011) it is also stated that operator actions are assumed to start after the first sequence of automatic actions, i.e. no grace time is assumed. According to BAREITH et al. (2003) and BAREITH (2007), the probability for the success of manual actions considered in the SPSA depends on the strength of the earthquake (expressed in terms of PGA values). However, the references do not provide any insight whether it was assumed that operator actions are successfully performed within short time periods (e.g. within 30 minutes) in the course of deterministic safety analyses.

The National report of Hungary (HAEA 2011) prepared for the EU Stress Tests mentions several seismic issues which are not fully resolved and should be further addressed. We appreciate this transparent approach of the Hungarian Atomic Energy Authority.

The safety significance of the different issues seems to be different. The biggest issue mentioned in HAEA (2011) is the potential for soil liquefaction, as it could act as an important initiator for common cause failures (CCF). According to HAEA (2011), the safety margin against soil liquefaction for pore water pressure causing the liquefaction in the layers between 10 and 20 m is only approxi-

---

[14] All figures and tables are presented in Annex 1.

[15] All figures and tables are presented in Annex 1.

mately 1.1 in case the assessment is done by simple empiric or semi-empiric methods. This margin is small compared to the possible far reaching consequences of soil liquefaction. Actual calculations resulted in a recurrence period of seismic events leading to soil liquefaction in the layers loaded by the main building of 14,000 to 18,000 years. This corresponds to an annual exceedance probability of 5.56×10^{-5} to 7.14×10^{-5} which is not insignificantly low. Therefore, adequate protective measure to avoid a total loss of relevant safety functions in case of soil liquefaction should, if necessary, be implemented rather quickly.

Experiences in Germany showed that anchor bolts to fix safety relevant equipment were mounted incorrectly. The problem was not that specifications of the bolts were incorrect. Instead, they were not sufficiently followed during the installation. This impaired the load bearing capacity of the bolts in case of an earthquake. Therefore, at some plants (e.g. Biblis NPP) large programmes were realised to replace anchor bolts and additional inspections were specified by the German reactor safety commission (RSK 2010). It is not clear whether comparable inspections of already installed anchor bolts have been performed in Paks NPP during the last years.

As pointed out in (HAEA 2011), the concurrent occurrence of an earthquake with plant states of short duration (e.g. displacement of the refuelling and hoisting machines during refuelling) was not assumed in the context of the deterministic safety case. According to HAEA, the contribution of such cases to the overall risk was evaluated in the probabilistic safety assessments. However, as no details are provided in HAEA (2011) no information is available concerning the question whether there is a potential for cliff edge effects. Recently, the German reactor safety commission recommended that superposition of operating conditions during low-power and shutdown operation of short duration with an earthquake should be considered to improve robustness. For the analysis of robustness, it is to be demonstrated that the design basis earthquake does not lead to significant impacts in the environment during temporary operating conditions of short duration (RSK 2012).

Based on the information contained in HAEA (2011) it can be concluded that a flood caused by postulated earthquake damage to the Slovakian hydroelectric power plant Gabcikovo does not impair the safety of Paks NPP.

3.3 Open questions/issues to be further addressed

Based on the information summarised in chapter 3.1 and on the evaluation in chapter 3.2 we recommend that the following issues should be further addressed in the bilateral process between Hungary and Austria. The listed issues are suggested to be treated in the framework of the project “Stress Tests Follow-up Actions” (topic 1):

- Reflection seismic data acquired during the seismic hazard assessment programmes in the late 1990ies and early 2000nds identified several Quaternary faults in the vicinity and near-region of the site. Have these faults been investigated with proper methodologies in order to constrain slip histories, youngest slip events, fault geometries and slip velocities?

- The information provided for the EU Stress Tests mentions that some paleo-seismological investigations have been carried out. Have these methods been applied in a systematic way to analyse the faults in the vicinity and near-region of the site and what are the results of these studies?
- SHA apparently includes earthquake recurrence models derived from fault parameters such as fault dimension and slip rate (models by Ove Arup). The Austrian experts would highly appreciate to get more detailed information on this issue. Have these models been applied to those Quaternary faults, which were identified by reflection seismic? What are the assumptions and input parameters for the fault models? Does the currently valid PSHA account for Ove Arup's modelling results?
- Current PSHA includes a logic tree approach, which attributes 10% probability to a model including active faults and 90% probability to "no faults". What is the justification for attributing such low probability to the active fault branch of the logic tree at the background of the existing evidence for Quaternary faults?
- The results of the assessments concerning necessity of measures envisaged to increase robustness of the plants against earthquakes – as presented in chapter 2.2.4. of the "National Report of Hungary on the Targeted Safety Re-assessment of Paks Nuclear Power Plant" (HAEA 2011) – and their respective implementation according to the "National Action Plan of Hungary on the implementation actions decided upon the lessons learned from the Fukushima Daiichi accident" (HAEA 2012), especially:
  - The results of further assessments concerning the potential impact of soil liquefaction and the respective implementation of additional measures to avoid CCF failure of vital safety functions. The Austrian side would highly appreciate to get information on the type of the safety relevant structures, systems and components endangered by liquefaction and on the measures envisaged to strengthen these SSCs.
  - The consequences of a potential failure of the three common demineralised water storage tanks of Installation II due to damages at the service building with respect to (secondary side) decay heat removal after an earthquake.
  - The results of the re-evaluation of the question seismic shutdown in the frame of the reconstruction project of the seismic instrumentation.
  - The potential for site flooding due to failure of pipelines of the main condenser cooling water system.
- The necessity to perform inspections of already installed anchor bolts to check whether they have been mounted correctly.
- Results of analyses concerning the possible consequences of a superposition of operating conditions during low-power and shutdown operation of short duration with a design basis earthquake.
- It should be clarified whether the deterministic safety case relies on successful operator actions within short time periods after an earthquake (e.g. within 30 minutes).

4 REACTOR PRESSURE VESSEL (RPV)

The reactor pressure vessel (RPV) is the central component of a nuclear power plant. It contains the reactor core. During operation, it 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 – 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 (ECC) water during an incident, which leads to cooling of the vessel wall (pressurised thermal shock, PTS). The failure of the pressure vessel constitutes a beyond design basis accident (BDBA) for all light water reactors.

In order to predict the progress of embrittlement during the lifetime of a nuclear power plant, a surveillance programme is performed: Samples of material are irradiated at higher neutron fluence rates than the vessel wall. For example, if the fluence rate is 10 times higher, the sample will in one year receive the same neutron fluence as the wall in 10 years (lead factor of 10). It is important that the samples are representative for the wall material, and the conditions of irradiation correspond to conditions at the wall (in particular, regarding temperature). Furthermore, there are indications that for the same overall fluence, the impact of a lower fluence rate over a longer time is higher than the impact of a higher fluence rate over shorter time (fluence rate effect). This can lead to non-conservative predictions of embrittlement.

The permissible extent of embrittlement is determined in PTS-analyses. For each scenario from a bounding spectrum of design basis accidents (DBAs), the conditions of the pressure vessel (internal pressure, temperature field in wall) and the stresses in the wall region close to the core are determined. On this basis, a fracture mechanical analysis is performed: For appropriately and conservatively selected crack postulates, stress intensity at the crack front is calculated and it is determined at which embrittlement state crack growth without arrest will occur. The value of the temperature (T_k) corresponding to this state is the critical value for the scenario. For each accident scenario, the critical T_k values are then compared; the lowest value of those, minus a safety margin, is the overall critical value and determines the permissible upper limit of embrittlement. Considering the neutron fluence rates, it can then be checked whether this value will be reached during the planned lifetime of the plant.

Increasing the reactor power (power uprate) tends to have the effect of increasing the neutron fluence rate in the vessel wall. However, this can be counteracted by an appropriate configuration in the core (low-leakage-core; for example by placing fuel with the highest burn-up in the outer positions).

4.1 Summary of information provided

PTS-Analyses

General

In the Environmental Impact Study of 2006, results of PTS-analyses for Loviisa NPP in Finland (also a VVER-440) were presented. It was claimed that the critical T_k for Loviisa (140 °C) would not be approached for the base material of Paks RPVs; weld material, however, could come close to this value after 50 years of operation (EIS 2006).

Measures to reduce the load during an accident sequence were briefly discussed (e.g. increasing the temperature of the emergency core cooling water and reducing the discharge head of the high pressure injection pumps), as well as the possibility of introducing a low-leakage-core.

The surveillance programme was mentioned, and it was announced that the values forecast for T_k were to be re-evaluated in the course of life time extension (LTE) licensing procedure, and that new PTS-analyses specific for Paks were to be performed to permit a re-evaluation and revised planning for countermeasures on this issue.

Additional information provided by Paks NPP further elaborated this point (ANSWERS 2006). It was emphasised that a complete new safety and component ageing analysis was required for licensing the lifetime extension. A full manufacturing database, a surveillance programme, a material ageing database, periodic non-destructive testing covering all relevant parts of the RPV as well as a complete new set of PTS-analyses were to be demonstrated in this context.

The thermo-hydraulic transients constituting the basis of those analyses were already identified and modelled in 2006. The full PTS-analyses had not yet been performed. On the basis of preliminary investigations, it was claimed that there were no obstacles identified so far to extending the lifetime by 20 years.

Brief information on the original status of the reactor pressure vessels, as well as on the surveillance programmes, was also given in EIS (2006) and ANSWERS (2006). It was stated that the surveillance programme at Paks NPP was designed in the same way as at Loviisa NPP. Inside each reactor of Paks NPP, six original sets of specimen were placed. The neutron fluence rate at the location of those specimens is 12–19 times larger than the one affecting the inner surface of the reactor wall.

Regarding power uprating, it was recognised that this could lead to an increase of the neutron fluence rate in the RPV wall. However, it was 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 would actually decrease.

In the summary of PTS calculations from the year 2008, the approach of the PTS analyses performed for Paks NPP was described (PTS 2008). This paper provided more details on geometrical data used in the analyses, neutron fluence calculations, surveillance programmes, thermal hydraulic analyses etc.

Regarding the safety margin between the T_k used in PTS-analyses and the T_k actually reached, specimens of the surveillance programme at Paks NPP have given indication that there is a margin of 25° between the actual T_k and the value of T_k used in PTS-analyses. In the IAEA "Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants", a safety factor of 10° is recommended.

Surveillance Programme at Paks NPP

The VVER-440/213 surveillance system and the extension of the surveillance programme at Paks NPP were elaborated.

The surveillance sets of the VVER-440 reactors are located in pipes welded to the core barrel. The specimens are placed into small capsules, and 20–22 capsules are connected together as a flexible chain. Every set of specimens consists of two chains. Every capsule contains two Charpy specimens or six tensile specimens. Six complete sets are located in the reactor. Every set includes 12 Charpy, six tensile and 12 Charpy size pre-cracked tensile specimens (for fracture toughness testing) from the base material, from the weld and from the heat affected zone. Two sets also have specimens over the core for monitoring the thermal ageing.

In VVER-440-s the specimens of the surveillance system are located in accelerated irradiation positions. In PTS (2008) it was stated that such accelerated surveillance system has the disadvantage that the operational changes (like use of low leakage core, or change of fuel type etc.) are not monitored. To eliminate this disadvantage, new specimen sets, with each set consisting of three forging materials[16], have been loaded in every unit of Paks NPP. Every specimen set consists of 12 to 16 Charpy specimens and up to 6 tensile specimens (smooth and notched) of each of the materials mentioned previously. Some of the Charpy specimens in each set are at the end of the chain in low flux region. After collecting these specimens from each unit they will be used to verify that no fluence rate effect occurs in the VVER-440/213 surveillance.

Evaluation of the surveillance data showed that the maximum value of T_{ki} (the transition temperature for initial conditions, i.e. unirradiated) after 60 years of operation is 111 °C for welds of Paks NPP units. With consideration of the applied safety margin for welds (δT_M = 16 °C), the maximum calculated T_{ki} is 127 °C. This value is far below the US-NRC criteria of 175 °C. For the base material, the value of maximum T_{ki} after 60 years of operation (incl. power uprate) is 89 °C. With consideration of the applied safety margin for base metals (δT_M = 15 °), the maximum calculated T_{ki} for base metals is 104 °C.

Neutron fluence calculations and dose rate effect

Regarding neutron fluence calculations, the scenarios and assumptions used for the calculations and results of the calculations were presented in PTS (2008). The results of neutron fluence calculations contained detailed neutron

[16] a special heat of 15H2MFA material, steel JRQ referenced by IAEA, and the original archive material of every unit using reconstituted specimens made from the remnants of unirradiated specimens

fluence distributions in the RPV walls from the 1st up to 60th operation year of the reactors. It was concluded in the paper that power uprating will not increase neutron fluence values compared to original core design and original power level. It was also stated that results of neutron fluence calculations for the surveillance specimen chains and results of detected neutron fluence values from the surveillance programmes are in agreement with the conclusion.

In PTS (2008) it was also elaborated that some specimen will be placed in a low flux region as part of the extension of the surveillance programme, to verify that there is no dose rate effect. The formulation appears to indicate that these specimens will allow direct control of dose rate effect. But this was negated by the Hungarian delegation at the 14th bilateral meeting in 2008. It was stated that there is no such direct control, and the statements on the dose rate effect are derived from information on impurities and thermal ageing.

The VVER-440 surveillance sets include thermal ageing specimens. At every unit of Paks NPP, a set of these specimens has been tested after four years of operation. It was stated that the dose rate effect is created by an interaction of irradiation embrittlement and thermal ageing – during short term irradiation there is not sufficient time to finish the thermal ageing process, yielding non-conservative embrittlement data. However, thermal ageing of the RPV steels at Paks NPP is negligible since they are of high quality (low phosphorus content), and because of that, it is argued that the dose rate effect can be excluded.

Impurities can also be a factor leading to a dose rate effect. In principle, the formation of copper-rich precipitates (CRP) can contribute to embrittlement. The rate of formation of CRP can be relatively higher at lower neutron fluxes: At high fluxes, unstable matrix defects (UMD) are formed which reduce radiation-induced diffusion and hence, CRP development. Thus, there can be a dose rate effect at copper concentrations of 0.1% or above. The effect will depend on the fluences involved, and on other impurities (Ni, Mn). At the 14th bilateral meeting in 2008, it was presented that the copper content of base material and welds varies between <0.1% and <0.3%. This point was not further discussed; the information provided indicates that a dose rate effect because of the copper content cannot be excluded.

In-Service Inspection programme

In-Service Inspection (ISI) programme were elaborated in PTS (2008). Paks NPP has an extensive In-Service Inspection (ISI) programme. The reactor pressure vessels are monitored from the internal surface by ultrasonic (UT) and eddy-current examinations. The inspection covers all components of the RPV which are relevant for PTS-analyses. Results of the ISI programme showed that integrity of the cladding is justified based on the examinations made so far, and under-cladding cracks exist only in the base metal and welds of vessels; they do not contact with the fusion surface (interface) of the base metal and the cladding, they are very small (the Through Wall Extent (TWE) is below 6 mm), and all cracks are in stable condition. The detection limit of the UT examinations for under-cladding cracks was not fully explained; it remains unclear whether it is 6 mm.

Thermal-hydraulic analyses

Specific thermal-hydraulic analyses were performed for the PTS evaluations with PTS-specific conservative assumptions (focused on maximum cool down rate, non-uniform cooling, minimum final temperature, maximum pressure and flow stagnation in the down-comer) according to the HAEA 3.17 Guidelines. Reactor system thermal-hydraulic analyses of reactor system were performed using the RELAP5/ATHLET computer codes with a detailed 6-loop input model of Paks NPP. The behaviour of the hermetic confinement was investigated by CONTAIN code for LOCA accidents. The detailed mixing analyses were performed for all non-symmetric cases using RE-MIX/NEWMIX codes. All performed analyses were reactor system thermal-hydraulic calculations. Their aim was to model both primary and secondary circuits including emergency core cooling systems. For the analyses, thermal-hydraulic codes were used with a detailed 6-loop model of Paks NPP. Primary and secondary circuits were modelled in detail according to the actual NPP configuration. The operator actions according to the Emergency Operating Procedures (EOP) were taken into account for some PTS transients. For the other transients with flow stagnation and thermal stratification in cold legs and predominantly single phase conditions in the down comer, the REMIX or NEWMIX code modified for Paks NPP was used.

The selection of PTS initiating events for thermal hydraulic analyses used a cut-off criterion; events with a frequency lower than 10^{-5}/year were not included which is in accordance with HAEA Guideline 3.17. There are some events listed in VERLIFE App. VI which were not included in the PTS-analyses (e.g. feedwater line breaks and low-temperature-overpressure (LTOP) events).

Adaptation of ASME BPVC Sections III and XI

At the 15th bilateral meeting in 2009, adaption of ASME code sections at Paks NPP was discussed. Although the ASME boiler and pressure vessel code (ASME BPVC) were not applied in the design and construction of VVERs, it seems worthwhile to use at least parts of the code if US materials could be connected to Russian materials, according to the Hungarian delegation. So, ASME code sections III and XI were to be applied to Paks NPP components. In the biannual HAEA report on “Recent Developments in Nuclear Safety in Hungary” of April 2011 (HAEA 2011a), it was stated that related ASME Standards and Codes will be published as Hungarian National Standards (in Hungarian language only) for the purpose of NPP pressure vessel and primary-secondary circuit, operation of passive-active mechanical components, in-service inspection, testing, maintenance and qualification. According to the report, the standard was planned to be published by the end of 2012 (HAEA 2011a).

At the 18th bilateral meeting in December 2012, it was stated that the adaptation of ASME BPVC has been completed.

Neutron flux on RPV wall and critical welding

Neutron flux on RPV wall and critical welding were discussed at the 17th bilateral meeting in 2011. In the early 2000s before new fuel elements were introduced, there was a problem with fuel and the best reload design could not be used. This resulted in a higher flux on the RPV wall during this time. The new fuel elements are extended in total effective length, therefore the flux maximum is closer to the welding joint. The comparison of predicted flux with earlier values showed differences which were caused not only by the fuel itself but also came from the new calculation model. There are no explicit limits for neutron flux, but there are limits for RPV parameters which are also influenced by the flux. Some uncertainties have been reduced due to the use of the more developed calculation model.

4.2 Evaluation and conclusions

Most of the questions and open issues regarding PTS-analyses have been clarified. Details on neutron fluence calculation as well as the surveillance system at Paks NPP and its extension were provided. However, there are some issues which have not yet been cleared.

Concerning the database for un-irradiated material, it was briefly mentioned in PTS (2008) that all un-irradiated data has been collected in a database and the reports are evaluated by independent experts and stored at Paks NPP. But the scope of the recording of properties still remains unclear. No other information on this issue was provided in the course of the following years.

Concerning safety margins of the ductile-to-brittle-transition temperature T_k in PTS-analyses, it was explained in PTS (2008) that the specimens of surveillance programmes at Paks NPP have shown better values of T_k than the values used in safety analyses (incl. PTS), with a difference of 25 °C. This could be taken as an indication that there is a safety margin of 25 °C for T_k in the PTS-analyses, which would be sufficient. However, this is an important issue and still has to be confirmed.

It is still unclear how far the dose rate effect caused by copper-rich precipitates (CRP) has been taken into consideration at Paks NPP. Also, the detection limit of UT examinations for under-cladding cracks is not completely clear.

As mentioned above, it was stated in PTS (2008) that the surveillance system of VVER-440, in which the specimens are located in accelerated irradiation positions, has the disadvantage that operational changes are not monitored, and that new specimen sets have been loaded in every unit of Paks NPP to eliminate this disadvantage. However, there was no further explanation of how operational changes can be monitored with the help of the specimens of the surveillance programme at Paks NPP.

4.3 Open questions/issues to be further addressed

The following questions are of interest to Austria and could be further discussed, as appropriate, at the regular bilateral meetings:

- Regarding the database of un-irradiated material, what is the scope of the recording of properties?

- Does the difference of 25 °C (mentioned in PTS (2008) Chapter 4.1.1) represent the safety margin for T_k in the PTS analyses?
- Has consideration been given to a dose rate effect caused by copper-rich precipitates (CRP)?
- Regarding the ISI programme, what is the limit of detection for ultrasonic (UT) examinations for under-cladding cracks (smallest crack depth which can be detected with certainty)? Do the 6 mm mentioned represent the detection limit?
- Please provide more elaboration on the method for monitoring operational changes by using specimens of the surveillance programme.

5 POWER UPRATE AND FUEL DEVELOPMENT

Increasing the electric capacity of a nuclear power plant beyond the original design value is generally referred to as power uprating (PU). 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 PU by increasing reactor power, the risk of plant operation can be increased. Margins relevant for safety might be reduced and plant ageing is accelerated.

With regard to the level of power increase, PU can be divided in three categories:

- Smaller PU (up to 2%); it can generally 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 PU (up to 7%); it usually involves changes to instrumentation set points, but still do not require major plant modifications.
- Extended PU (up to 2%); it may require significant modifications to major balance-of-plant (BOP) equipment. BOP is the summary of all components and systems in the plant that are needed for harmonious, safe and efficient operation.

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 overall 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 PU.

In order to assess the feasibility of a thermal PU, plant behaviour 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 system.

PU 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 (see also chapter 4).

5.1 Summary of information provided

In the Preliminary Environmental Study of 2004 (PES 2004), it was mentioned that PU to a nominal power of 500 MW_e was planned 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.

In the Environmental Impact Study of 2006 (EIS 2006), it was stated that, according to the results of a feasibility study from 2001, development of a new type of fuel is required to achieve the planned 108% power increase. The new fuel was to be developed in two phases.

Fuel development

The two phases of fuel development were described in more detail in a report paper from 2009 by Larisza Szöke (Paks NPP) (SZOEKE 2009), provided by the Hungarian side in the course of the Roadmap. In the first phase, the grid division of the rods of the working cartridge was changed from 12.2 mm to 12.3 mm in order to guarantee a more effective cooling. A neutron absorbing hafnium plate was installed in the upper section of the fuel parts of the control and safety cartridges so that axial neutron-flux and performance distribution become more even and the released heat is distributed radially. The introduction of this modified, temporary fuel is sufficient to achieve the 108% power level. However, in order to provide the excess reactivity, more new cartridges than before are necessary during the transfers so the fuel cycle will become less economical. The second stage of the development is to optimise the fuel management through the improvement of the fuel. With a completely new type of fuel it is possible to accomplish a five-year cycle which is more economical than the fuel-cycle using the old fuel. In order to reach this goal, the enrichment must be enhanced and out-burning poisons must be used. The average enrichment of the optimised cartridge is 4.20% with three absorber rods containing an out-burning poison (gadolinium).

After completion of the PU, a programme to develop new type of fuel was carried out (second phase of the fuel development). At the 17th bilateral meeting in 2011, it was explained as the reason for fuel development that burn-up of the fuel decreases as a result of PU. Due to its higher initial enrichment, higher burn-up is possible with the use of new fuel. Information on fuel geometry and results of test programme, which was carried out before the introduction of new fuel, were presented in the meeting. The use of this new fuel is first tested in unit 4, and then the fuel will be introduced to other units step by step. In the presentation it was stated that PU and the connected fuel development has been finished successfully. Accuracy of reload design calculations for new fuel is satisfactory, and the new fuel can be introduced to all units.

In JÁVOR (2009), more details on the fuel development were presented. The new fuel has an average uranium enrichment of 4.2%. Due to the higher enrichment, there were two problems to be resolved: One is to guarantee subcriticality during the delivery and storage; the other is the high k-infinitive (neutron multiplication factor) at the beginning of the cycle. Gadolinium (Gd) has a large neutron absorption cross section. The Gd content in a pin is 3.35% and it will burn up till end of the first cycle of fuel. Six Gd pins should be in a fuel element because of 60 degrees symmetry. In this case k-effective has a problematic behaviour at the beginning of the cycle. Three pins are used to moderate this effect. Fuel elements are rotated by 60 degrees in every second sector. The enrichment was reduced around the corner to reduce the power peak there. The uranium enrichment of pins with Gd is 4.4% and the enrichment of pins at the corner is 3.6%. All the other pins have 4.0% enrichment.

Power uprate

Safety and operational limit

According to EIS (2006), results of analyses have demonstrated that PU would not lead to a decisive reduction of safety margins to the limits. It was also mentioned that, according to a feasibility study made by VEIKI AG concerning the effects of a PU on the ageing processes of the main components of the units, PU would accelerate ageing processes, but it was emphasised that the LTE would not be significantly influenced by this and that the effects can be minimised by means which either are already or were going to be implemented. As a pre-requisite of licensing of the PU required by the HAEA, the same level of safety as before had to be maintained.

At the 16th bilateral meeting in 2010, experiences of the PU at Paks NPP were presented. Necessary modifications were categorised based on the reasons (process technology, keeping operational limits, or preserving safety margins). Categorised as modifications to keep the operational limits: introduction of new fuel type, reconstruction of primary pressure control system and modernisation of in-core monitoring system. Categorised as modifications for preserving safety margins are the up-scaling of some of the trip signals, change of parameters of the HA, and increment of boron concentration in the primary circuit to 13.5 g/kg.

Results of safety analyses were presented. It was stated that all DBA initiating events were recalculated for the new power level. Since the PU was compensated by plant modifications, there is neither a significant reduction of any safety margin, nor relaxation of any acceptance criteria or operational limits. The impact of PU on large release frequencies is smaller than the uncertainties in the calculation of the large release frequencies.

The decisive parameters for which reserves have to be preserved are the maximum sub-channel outlet temperature, the linear heat rate, the fuel rod power and the fuel assembly power. In the presentation, tables were shown with the values of these parameters for different fuel positions. Also, the reserves to the operational limits were presented. The uncertainties of the determination of these reserves were shown to be considerable; for three of the four parameters mentioned, the uncertainty was more than half of the reserve. Still higher uncertainties were presented for the reserves to fuel burn-up limits.

Further explanations regarding definition of limits, reserves and safety (engineering) factors were provided in a presentation at the 17th bilateral meeting. For important parameters, there is a reduced limit (operational limit), and a (higher) limit derived from safety analyses. The 'reserve to limit' (values of which were presented by Mr Elter at the 16th bilateral meeting (2010)) refers to the reduced limit. This reserve to limit has decreased because of the power uprate.

The difference between the reduced limit and the limit derived from safety analyses constitutes a safety factor or safety margin (to cover uncertainties, e.g. from measurements, tolerances from fuel fabrication as well as small variations in geometry, enrichment etc between fuel assemblies). This safety factor has not decreased after the power uprate.

Influence of PU on RPV and SGs

There is a more detailed discussion in EIS (2006) concerning the reactor pressure vessel (RPV) and the steam generators (SGs). For the SGs, 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 SG tubes.

Regarding the RPV, it is recognised in EIS (2006) that PU could, in principle, lead to an increase of the neutron flux in the RPV 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 chapter 4).

Influence of PU on containment behaviour and BDBA events

Results of safety analyses assessing the effects of PU in case of design basis accidents were also mentioned in EIS (2006). The analyses included calculation of the radioactive emissions and the resulting doses in case of design basis accidents for the power level before PU (100%) and for the increased power (108%) to allow comparison. The following results were regarded as most important and were reported in EIS (2006):

- Thermo-hydraulic analyses demonstrated that in case of a large-break loss of coolant accident (LBLOCA) the integrated mass and energy flow is higher for 100% than for 108% power, due to the modifications of the hydro-accumulators. The initial hydro-accumulator pressure was decreased from 58.8 to 35 bar, with a simultaneous increase of their inventory by 10 m^3. A similar modification which has been already performed at the Loviisa and Dukovany NPPs was reported to have a clear positive safety effect: There would be better cooling in case of LBLOCA scenarios.
- Maximum pressure inside the containment is higher for 100% power than for 108% power.

After PU, 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 was reported that the analyses show that this modification indeed achieves this purpose and leads to lower surface temperatures. Further modifications and its expected effects were also explained in EIS (2006); these include modifications of the pressure control system, modifications of main coolant pumps, increasing boron concentration in the primary circuit, and improvements in reactor zone monitoring.

In ANSWERS (2006), it was stated that the amount of radioactive materials potentially released during accidents will decrease.

At the 15th bilateral meeting in 2009, it was stated in the presentation concerning influence of PU on containment behaviour and BDBA events at Paks NPP that the PU to 108% had been performed for all units. PU for unit 4 was achieved in September 2006, unit 1 in July 2007, unit 2 in December 2008, and unit 3 in November 2009. The obtained steps for PU were listed. Advanced fuel with different core loading pattern design has been utilised. Minimal core flow rate was increased from 39,450 m³/h to 40,300 m³/h. Stabilisation of the primary pressure had been carried out (narrowing of the allowed pressure range by the new regulator system on pressuriser). Critical boron concentration at BOC condition was increased from 12 g/kg to 13.5 g/kg. Hydro-accumulator (HA) pressure was reduced, as already mentioned, from 58.8 to 35 bar.

In the same presentation, it was stated that the effect of PU on containment pressure is negligible due to HA-pressure reduction. LBLOCA (maximal DBA case) produces the highest containment pressure (a figure demonstrating the progression of primary mass inventory with both HA pressures before and after the reduction was provided). In the case of the higher HA pressure (58.8 bar), injection from the HAs starts already after 6 to 8 seconds, and a significant part of the HA water is lost via the break. With the reduced HA pressure (35 bar), HA injection is delayed to 20 – 25 seconds, so the amount of water loss is reduced. Peak of containment pressure is reached earlier than the start of HA injection.

Regarding the influence of PU on PSA, it was stated in the presentation that impact of PU on core damage frequency (CDF) had been quantitatively evaluated. The calculated CDF remains practically unchanged. In SZOEKE (2009), it is mentioned that the CDF of the unit by nominal performance operation in the state of 108%, considering the effects of related modifications is 1.09 x 10^{-5}/year (about 1.4% higher than the CDF before PU). Main issues which potentially could be influenced by the PU are the success criteria of decay heat removal and the time available for operator actions. According to the presentation, the heat removal success criteria are not affected, and therefore, the core thermal and hydraulic limitations remain unchanged. As a positive effect of the HA pressure reduction in relation to BDBA situations, it was mentioned that the time until overheating in case of inadequate core cooling (i.e. SBLOCA without successful HP injection) increases. Hence, there is a higher chance for successful operator actions to re-establish high pressure cooling and to carry out depressurisation of the primary system allowing injection of low pressure water sources. It was mentioned that the available time frames for successful operator response in other cases are not significantly reduced.

Concerning the impact of PU on large release of fission products and progression of severe accident, the analyses were performed on the basis of level 2 PSA calculations. Radioactive inventory increased due to the PU, and there are eventual changes in source terms (for full power and shutdown states). Diagrams comparing the release of caesium and other isotopes to the environment for 100% and 108% power level were presented. For full power states, the changes are partly positive and partly negative. No significant effect of the PU in this regard could be established. For shutdown states, the release of uranium is higher due to PU, but the releases of caesium and iodine are lower. So it was also concluded that the PU has no significant effect to the release in shutdown states.

There are no changes in the containment event trees caused by PU, but there are some changes in the frequencies due to HA pressure reduction. Results of hydrogen (H_2) production analyses showed that integrated mass of produced H_2 after PU is higher, but the production rate is lower. The sequence being used for the analyses is a small break loss of coolant accident (SBLOCA) without high-pressure safety injection, with primary depressurisation. It was concluded that, with regard to hydrogen management, the situation would be easier to manage for 108% power level. It was also mentioned that the efficiency of passive autocatalytic recombiners (PAR) is higher after the PU. It was also mentioned at the meeting that the time frame for depressurisation of primary system is about 1–2 hours, and time until core uncovery in case of the SBLOCA without HP injection is about 40 hours.

Regarding SAM strategy and mitigative actions, it was stated that proposed mitigative SAM measures remained unchanged. A short table presenting SAM mitigation action and its evaluation in relation to the effect of PU was provided. Mainly, it was stating that the proposed mitigative actions are applicable and adequate, and in some points only needed small changes or that the changes of timing are still within the uncertainty bandwidth.

With regard to the reconstruction of the primary pressure control system, it was said at the 16th bilateral meeting that it is vital for safe operation that a distance to the saturation temperature is kept. The saturation temperature depends on primary pressure. Core outlet temperatures increase due to the PU, decreasing the distance to saturation temperature. This is compensated for by a more accurate primary pressure control, permitting a more accurate determination of saturation temperature at every moment. In brief: The distance to saturation is decreased, but can be determined and controlled with greater accuracy.

Concerning the modernisation of the in-core monitoring system, it was explained that improved in-core monitoring and improved processing capability of the calculation system allow more accurate determination of vital parameters, in particular the sub-channel outlet temperatures. Also, the mesh of values over the core, which are obtained partly by measuring and partly by calculation, has become finer after the modernisation.

Vibration of secondary pipelines

Regarding the vibration of secondary pipelines, it was explained in the written response to questions from the Austrian side that vibration velocity measurements have been performed before PU, according to ASME-OM-SG-2000 Code. Measurements after PU with power level of 104% and 108% were carried out at the same places. For each unit and power level an average value was calculated and compared to the original power level.

The vibration rate was below the ASME limits in all cases. Three diagrams are presented with the conclusion that there is practically no change of vibration velocity in the feed water system, because of PU, and that acceptance criteria are fulfilled. However, the diagrams and its following conclusion were not completely comprehensibly, because the Y-axis in the diagrams has no scale. For the other systems it is stated that the increase is variable but still inside the acceptance criteria.

Concerning the uncertainties of the vibration measurements, the Hungarian response explains that they have used standard measurement equipment. The uncertainties are not specified but technical data of the equipment is given and the Hungarian experts concluded that the uncertainties due to the measurement are not relevant.

5.2 Evaluation and conclusions

In the course of the Roadmap, many questions have been clarified, a significant amount of information regarding the new fuel, safety factors for reactor operation, changes of safety systems and the result of accident analyses has been provided. However, there are still some points which have to be clarified.

In the presentation concerning the influence of PU on PSA at the 15th bilateral meeting, it was mentioned that the time until overheating in case of SBLOCA without successful HP injection is longer than a positive effect of HA pressure reduction. However, there was no quantitative information about how long is the time until overheating for 100% and 108% power level. It was also mentioned that the available time frames for successful operator response in other cases are not significantly reduced, but no quantitative information was given.

Regarding the containment, it was stated at 15th bilateral meeting that the results of hydrogen (H_2) production analyses showed that the integrated mass of produced H_2 after PU is higher, but the production rate is lower. And it was concluded that the situation for 108% power level would be easier to manage with regard to hydrogen management. However, it was not further explained, how significant the changes on the mass of produced H_2 and the production rate compared to the values for 100% power level are.

Regarding SAM strategy and mitigative actions, a short table was presented at the 15th bilateral meeting, showing SAM goals, the related mitigative actions and comments or evaluation results in relation to PU. Mainly, it was stated in the table that the proposed mitigative actions are applicable and adequate, and in some points only needed small changes or that the changes of timing are still within the uncertainty limit. However, there are no further explanations on background information which led to the comments and evaluation results presented in the table.

It was explained that the results of safety analyses have shown that there is neither a significant reduction of any safety margin, nor relaxation of any acceptance criteria and operational limits. IAEA publication No. NP-T-3.9 "Power Uprate in Nuclear Power Plants: Guidelines and Experience" (IAEA 2011), emphasises the importance of the establishment of a trending programme to track and review the trending of parameters in order to ensure that the actual results are comparable to the predicted results in term of margin reduction. It is also stated in IAEA (2011) that "areas that do not compare favourably to the predicted results are candidates for further review to determine and understand the discrepancy" and that "corrective actions to resolve these discrepancies must be implemented". Information on the results of such a trending programme at Paks NPP would be of interest to the Austrian side.

5.3 Open questions/issues to be further addressed

Of the following questions/issues, all but the last one will be treated in the framework of the above mentioned project "Stress Test Follow-up Actions", topic 3. The last issue could be discussed, as appropriate, at the regular bilateral meetings.

- In case of SBLOCA without successful HP injection, how long is the time until overheating for the power level after PU and for previous power level?
- Quantitative information on available time frames for successful operator actions for other cases (both for 100% and 108% power level) would still be of interest to the Austrian side.
- Quantitative information on the changes in the integrated mass of produced H_2 and the production rate after PU would still be welcome.
- More information on the evaluation of SAM mitigative actions in relation to PU would provide better understanding.
- Information on the results of the programme to track and review the trending of parameters after PU would be appreciated.

6 CONFINEMENT SYSTEM AND BDBA

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

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

The confinement systems of VVER-440/213 plants have been designed with relatively high leak rates, compared to Western PWRs. 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. However, these investigations and tests concerned solely design basis accidents (DBA). Regarding protection against severe accidents (BDBAs), it is noteworthy that the original containment capability to limit releases appears to be somewhat inferior to Western PWR containments (WENRA 2000).

6.1 Summary of information provided

Containment structure

Before the 14th bilateral meeting, the Hungarian side provided the paper “Containment behaviour during DBA and BDBA events at the Paks NPP”, which is referred to as CONTAINMENT (2008) in the following.

The Paks containment structure consists of two major features: the steam generator compartment and the bubbler tower that features bubbler-condensers for accident steam suppression (see figure[17] 6-1). In the event of an accident resulting in elevated pressure inside the containment, steam-air mixture flows through a steam corridor to the bubbler tower and through the condenser water trays, resulting in condensation of the steam. The non-condensed gas (primarily air and hydrogen) is routed from above the trays to the air traps through one-way valves. In the containment, steam is condensed by the containment spray systems. The water trays are supported by steel beams, at 12 elevations, connecting the wall. These beams are also important in providing restraint for the wall, significantly increasing internal pressure capacity of this wall.

The reinforced concrete wall and slab panels of the containment have steel plate liners either on the inside surface or embedded into the panel close to the outer surface of the panel. Except in the bubbler tower, in the wall and slab panels that have the 6 mm thick hermetic boundary steel plate, there is also a

[17] All figures and tables are presented in Annex 1.

4 mm thick steel plate on the opposite side of the panel. A nominally 2 m thick reinforced concrete base slab supports the containment structure. The main features of the outer pressure boundary, consisting of the steam generator compartment outer walls, floor and roof, the corridor connecting the steam generator compartment to the tower, and the tower have rectangular geometry patterns.

It is stated, while a cylindrical containment wall of the typical US and Western European containment designs resists the pressure loading in membrane tension with generally negligible bending, the flat rectangular panels of the Paks containment pressure boundary are subject to significant bending moments (CONTAINMENT 2008). At the 14^{th} bilateral meeting in 2008, it was added that the wall thickness of the concrete structure of the containment is between 0.5 m–2 m.

Ageing Management

Ageing effects could weaken the integrity of the confinement system. The Preliminary Environmental Study and the Environmental Impact Study and contain only little information about ageing of the confinement system (PES 2004; EIS 2006).

The ANSWERS (2006) provided by the Hungarian side give more information: It is reported that the construction is monitored and inspected during the operation on the basis of status control and maintenance (ageing management) programmes. The types and extent of the experienced and expected ageing and deterioration processes 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 proven and practically tested methods. It is stated that no deterioration of reinforced concrete and concrete steel structures has been experienced so far.

Furthermore, 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.). Regarding the state of the containment and the main building, it was stated that, following the ageing management, status control and maintenance programmes 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 (LTE), licensing the overall review of the ageing management programmes relating to constructional components has been performed (ANSWERS 2006).

Leakage rate

According to reference document, the design leakage rate is 14.7 vol% per day for the maximum DBA case (LBLOCA). During DBA events the design pressure of 250 kPa will not be exceeded. Results of measurements of the leak rates at the four units during outage in 2008 are given. These values show a considerable variation (CONTAINMENT 2008):

- Unit 1: 8.6 vol%/day;
- Unit 2: 10.2 vol%/day;
- Unit 3: 4.8 vol%/day and
- Unit 4: 7.7 vol%/day.

At the 14th bilateral meeting it was clarified that the given values are derived from the test at 120 kPa, but extrapolated to the design pressure of 250 kPa. Only for unit 3, a full pressure test was already performed and showed a lower leakage rate than calculated. Full pressure tests at the other units were planned in the near future. It was also stated that differences in the leak rates are of no significance since all values are below the design limits. For the test the hermetic zone is closed, pressure is built up, and the decline in pressure is measured. The arrangements for temperature and pressure testing are prescribed by the authority. The air traps are included in the test, because they cannot be closed. According to the National Stress Tests report, the actual leakage rates are between 4–8 vol%/day at 1.5 bar overpressure (HAEA 2011).

Capacity of confinement system

In EIS (2006), the resulting confinement pressures for various design basis accidents (DBAs) are presented. 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 bubble condensers.

It is summarised in the ANSWERS (2006) provided by Paks NPP that in case of a possible accident, the safety level is maintained and the environment is protected by safety and localisation systems consisting of active and passive components.

The aim of the accident containment analyses is to determine the pressure load and leakage rate during design basis accidents (DBA). Therefore, two criteria have been demonstrated:

a. Calculated containment pressure during accident will not exceed the design containment pressure (250 kPa);
b. Pressure drop on bubble condenser will not exceed the design value (30 kPa). Containment behaviour during DBA events has been analysed by CONTAIN code in one day time interval, except main steam line break (MSLB) and feedwater line break (FWLB), which were calculated for 5,000 s (CONTAINMENT 2008)

Maximum pressure load (pressure difference) to containment walls is reached in max. DBA case (with max. ECCS configuration): 222.8 kPa, which is only about 10% lower than design pressure of containment 250 kPa. Maximum pressure load during medium and small break LOCA cases is much less than during LBLOCA case. Maximum pressure difference (20.1 kPa) is reached during LBLOCA case, but it is much less than the design value (30 kPa).

At the 14th bilateral meeting it was stated that the average pressure capacity of the confinement system is 500–600 kPa; significantly higher than the required 250 kPa. However, some weak points exist (elements in bubble tower) and the lowest value (375 kPa) determines the containment capacity.

As part of the Level 2 PSA, an evaluation of the capacity of the containment structure for elevated (BDBA and severe accident) temperature and pressure loadings was required. This evaluation was carried out by the ABS Consulting Inc. in 2001–2004. The evaluation methodology is based on estimating the capacity of the containment structure in terms of probabilistic parameters for a number of possible failure modes. The valuation of a rectangular structure under pressure is more complicated than for a cylinder, Paks NPP overpressure study required more extensive evaluations than the typical studies for cylindrical containment (CONTAINMENT 2008).

In CONTAINMENT (2008), a probabilistic description of pressure capacity is provided. The mean overall containment capacity was computed to be 350 kPa (50% failure probability) and the “high confidence of low probability of the failure” (HCLPF) capacity to be 235 kPa (5% failure probability).

A discussion of the behaviour 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 up-rated power level (108%). Thus, this issue is discussed in chapter 5: Power Uprate and Fuel Development.

Severe Accident Management

The reference document “Overview of Paks Severe Accident Management Status” (SAM 2009), provided by the Hungarian side before the 15th bilateral meeting, gives a general introduction concerning the function and elements of severe accident management guidelines (SAMGs). It is stated that the SAMG package was completed in May 2008. Before the usage of SAMGs is possible, however, a few technical modifications have to be implemented for some plant systems and equipment. The questions discussed at the 15th bilateral meeting focused on those technical modifications.

The National Stress Tests report provides comprehensive information on Severe Accident Management (SAM) at Paks NPP (HAEA 2011). This applies to the issue discussed during the Roadmap LTE Paks NPP as well as to issues which were highlighted by the severe accidents at Fukushima NPP: SAM for spent fuel pools, management of multi-unit severe accidents and site organisation for accident management.

As a pre-condition for the planned lifetime extension the nuclear authority required that the modifications necessary for the management of beyond design basis events and severe accidents shall be completed prior to the expiry of the original design lifetime of each given unit (HAEA 2011).

The modifications implemented at Paks NPP with regard to Severe Accident Management (SAM) are aimed to stop any assumed severe accident event sequence and to bring the unit to safe cold shutdown state. Two key elements are the execution of the technical modification belonging to SAM and the introduction of Severe Accident Management Guidelines (SAMG). The principal elements of severe accident management modifications are as follows (HAEA 2011):

- External cooling of the Reactor Pressure Vessel (RPV) by discharging water from the localisation tower and flooding the reactor cavity,
- Severe accident management measuring system,
- Severe accident diesel generators for supplying electrical power to SAM instruments,
- Hydrogen management under severe accident conditions by passive autocatalytic recombiners,
- Prevention of coolant loss from the spent fuel pool due to pipeline rupture.

All SAM modifications are implemented already at unit 1, and will be implemented at unit 2 until December 31, 2012, at unit 3 until December, 31, 2013 and at unit 4 until December 31, 2014 (HAEA 2011).

Topics dealt with during the Roadmap process

Depressurisation of primary system

Depressurisation of the primary system is an important step of severe accident management, in particular as a pre-condition for in-vessel retention of the molten core. According to SAM (2009), depressurisation is performed using pressuriser safety and relief valves. There are two safety valves and one relief valve. It is mentioned that an autonomous power supply for the pressuriser safety valves is to be installed. A preliminary license for implementation has been granted by the authorities.

At the 15th bilateral meeting it was stated that the existing valves are already fully qualified for all depressurisation scenarios. It was also explained that the new autonomous power supply is for the case of station blackout only (ca. 100 kW). Information concerning the number of valves required for depressurisation and for the time frames involved was not provided at the meeting, but it can be found in the papers received at the meeting (LAJTHA et al. 2009). The issues discussed at that time could be regarded as closed.

In the case of a total blackout and/or the loss of the ultimate heat sink, primary pressure is high in the early stage of the process. The reduction of the pressure is important because certain elements of the emergency core cooling system can start to operate only on lower pressure level. The Symptom-based Emergency Operating Procedures (EOPs) give instruction on unconditional pressure reduction above 550 °C core outlet temperature. If the core outlet temperature further increases during the application of the Symptom-based Operating Procedures and then exceeds the value of 800 °C in the case of total blackout, or the value of 1,100 °C in any other case, then the SAMGs have to be applied (HAEA 2011).

Reactor cavity flooding, external cooling of RPV

According to the reference paper, external cooling of the reactor pressure vessel in case of a core melt by flooding the reactor cavity is the decisive measure to control a core melt accident, to retain the core inside the reactor pressure vessel and to keep releases relatively low. The reference paper states that these measures are in preparation, together with the strengthening of the reactor cavity door (SAM 2009).

At the 15th bilateral meeting, it was explained that analyses with ATHLET and other codes have been performed to show that in-vessel retention is feasible, for two cases: 100% and 108% power. The calculations show that cooling by a circulation loop via the steam generator (SG) boxes will function even in situations with high temperatures. However, the calculations still required confirmation by experiment. In particular, this concerned the weak part of this strategy – there are very small gaps at the lower part of the reactor cavity, which limit the circulation. The CERES provided this confirmation; tests were performed during the following years (see below). There is cooperation in this field between Czech, Hungarian and Slovakian experts. It seems clear from information received at other meetings that the concepts pursued are very similar, and also similar to the concept implemented at Loviisa NPP. The questions concerning the comparison of in-vessel retention concepts in different VVER-440s (planned for Dukovany and Mochovce NPP, applied at Loviisa NPP) could not be discussed due to time constraints.

According to HAEA (2011), the external cooling of the reactor pressure vessel is already implemented at unit 1, and will be implemented during the main outages at unit 2 (2012), unit 3 (2013) and unit 4 (2014). According to the NAcP, no further action is necessary (HAEA 2012).

The National Stress Tests report provides more information on this topic: At first, the external cooling of the reactor pressure vessel requires water discharge from the localisation tower (from the bubble trays) to the floor of the containment, and then the water can be discharged to the reactor cavity from there by the force of gravity (see figure[18] 6-2). The appr. 1,180 m³ of water and the coolant from the primary circuit can be used to fill up the 270 m³ reactor cavity. The discharge of water from the localisation tower has to be started before the evolution of extended core damage, when the core outlet temperature reaches 550 °C. The discharge valves can be operated when the primary pressure is lower than 20 bars and the water level on the containment floor reaches a given level. The execution of the measures requires operator interventions pursuant to the "Water supply to the hermetic compartments, flooding of the reactor cavity" instruction. The electrical power of the discharge valves can be provided from the normal, safety and severe accident power supplies.

The water transfers the heat from the wall of the reactor pressure vessel (in the way of boiling and condensation) to the containment through natural circulation.

According to calculations made on the external cooling of the reactor pressure vessel, the intactness of the reactor pressure vessel can be maintained. The conclusions drawn from the calculations were justified by the modelling of the heat flux occurring during the accident and the actual geometry of the reactor cavity in the frame of CERES experimental analyses (HAEA 2011).

The information concerning CERES provided by the Hungarian side at the 18th bilateral meeting in December 2012, as well as the documents transmitted shortly afterwards, contain a wealth of information and prove that a consider-

[18] All figures and tables are presented in Annex 1.

able amount of relevant work has been performed. However, it cannot be excluded at the moment that some additional clarifications might be of interest to the Austrian side, for example regarding the correspondence of experimental results with results of calculations. This point at first has to be further examined by Austrian experts.

Prevention of overpressure of the containment

After the flooding of the reactor cavity, the residual heat of the molten core in the RPV warms up the coolant in the cavity. The evaporation of the coolant increases the quantity of steam in the containment; if the sprinkler system is not operable, the containment pressure will gradually increase.

The long-term evolution of pressure is highly dependent on the actual value of the containment leakage rate. The evolution of containment pressure, as a function of different leakage rates, is shown in figure[19] 6-3. According to HAEA (2011), the HCLPF value (3.35 bars absolute pressure) is the determinant with regard to avoid damage, and timeliness of the intervention. Depending on the leakage rate, the containment pressure exceeds this value within 3–8 days. If pressure reduction does not occur in the meantime, the containment pressure will increase until the containment will fail or until the mass flow leaking from the containment will be equal to the generated mass flow. Consequently, the relevant accident management guideline requires the reduction of the pressure in the containment. This can be achieved by cooling of the air volume or by relieving the containment pressure through the venting system. If the electrical power supply is totally lost, then the air volume cannot be cooled by design tools; the only possibility is to discharge air through the venting system of the containment. Unfiltered release can be executed only after the evacuation of the area around the nuclear power plant; thus further corrective measures are identified to manage the prevention of containment overpressure. Two concepts are on the table in this regard. One concept aims at the filtered discharge of the containment. Another concept aims at the long-term cooling of the containment that also handles the containment overpressure.

It is pointed out in the National Action Plan that Paks NPP prepared the latter concept for the implementation which recommends the installation of an active cooling system. The final deadline for this measure is December 15, 2018 (HAEA 2012).

According to HAEA (2011), damage to the reactor pressure vessel is not expected. But should vessel damage happen, then, in principle, it may occur in two situations: before flooding the reactor cavity and after it. In the first case, the experts of the Technical Support Centre (TSC) have to decide on whether the flooding of the reactor cavity after the damage to the vessel is to be performed, based on whether the debris can be cooled down. On the other hand, a steam explosion may occur if too much water is used. In the second case, if the reactor pressure vessel suffers damage after the flooding of the cavity, then a relatively small amount of molten fuel will escape and then the solidifying debris will block the route (HAEA 2011).

[19] All figures and tables are presented in Annex 1.

Measures against hydrogen hazards

In SAM (2009), it is stated that 30 passive hydrogen recombiners for severe accident situation are to be installed. A preliminary license for implementation had been granted by the authorities. The technological preparation was in progress at that time. At the 15th bilateral meeting, it was explained that the locations of the recombiners were not definitely determined yet. Most of them were planned to be installed in the steam generator boxes. In the air trap, there were to be one or two recombiners. It was also stated that some accident analyses relevant for the placing of recombiners had already been performed, using the computer codes GASFLOW and MAAP. Furthermore, Hungarian experts were participating in international projects which can provide useful information for the selection of the most efficient places.

The National Stress Tests report provides more information regarding the hydrogen hazard: Meanwhile the 60 (30 pairs) NIS type passive autocatalytic severe accident recombiners are installed in the containment at all units. The installation process of hydrogen recombiners was accelerated after the Fukushima accident (HAEA 2011).

According to HAEA (2011), the hydrogen generation due to zirconium water reaction can be so intensive in certain processes that, in spite of the recombiners, the hydrogen may burn during an initial short period of time. However, the concentration of hydrogen is low enough that the hydrogen burning cannot jeopardise the integrity of the containment. In a later phase of the severe accident process, the hydrogen concentration further decreases, and thus the gas mixture is not flammable anymore. Consequently, the relevant instructions of the SAMGs primarily focus on the monitoring of the hydrogen concentration.

The hydrogen leaks to the reactor hall and the technology building through the permissible leakage of the containment, but burnable gas composition cannot develop in these rooms, the hydrogen concentration remains below 1 vol% (HAEA 2011). However, further studies on hydrogen generation and distribution in the reactor hall are planned. HAEA requested to develop more detailed studies to determine the quantity and distribution of hydrogen in the reactor hall during an accident that simultaneously assumes two damaged spent fuel pools and two (one open and one closed) damaged reactors within a twin unit.

The available analysis results cannot fully exclude the evolution of flammable hydrogen concentration based on the quantity and distribution of hydrogen produced during simultaneous accidents of two spent fuel pools of an installation, an open reactor under refuelling and a closed one under operation. The reliable assessment of this issue requires less conservative, three-dimensional calculations. According to the NAcP these studies are finalised in 2012 (HAEA 2012).

Source terms, Effects of SAM strategies

At the 15th bilateral meeting, information on frequencies of different release categories and on source terms was provided. The question about the goal of the mitigation measures was posed: First answer from Paks NPP was that no exact release criterion for BDBA as in Finland is defined in Hungary, but might be defined later. HAEA argued that BDBA emission has to be as low as possible, but the Finnish target of 100 TBq cannot be kept, because of different types of containment. However, according to HAEA (2011), the introduction of severe accident management significantly reduces the probability of large radioactive releases; it is expected that this value will not exceed the more strict requirements for new-built units (see also chapter 5: Power Uprate and Fuel Development).

Containment bypass – steam generator tube/collector rupture

Accidents with the steam generator (SG) tube or collector rupture lead to particularly high releases, since the containment is bypassed. No technological measures are mentioned for this case in the reference paper (SAM 2009). Due to time constraints, this topic could not be addressed at the 15th bilateral meeting or at later meetings.

In a publication in the same year (ELTER 2009), a new PRISE (primary to secondary side leakage) strategy is briefly mentioned: Bleed from ruptured SG to the containment before it is filled up. It is stated that this measure is “implementing now”. No detailed information is provided.

Additional relevant topics from the EU Stress Tests

The following information is based on the National Report of Hungary on the Targeted Safety Re-assessment of Paks Nuclear Power Plant (HAEA 2011) and the recently published National Action Plan (NAcP) (HAEA 2012).

Measuring and control instrumentation

The severe accident measurement system is an important element of SAM, because certain parameters are required to be known for the execution of interventions defined in the SAMGs. The construction of the measurement system guarantees its operability under severe accident conditions (temperature, radiation, humidity). Batteries can supply electrical power to the measurement system for 3.5 hours. According to HAEA (2011), this period is sufficient to put the severe accident diesel generators, which supply electrical power to the measurement system, into operation and to start them. According to the National Action Plan the installation of a hydrogen monitoring system as part of the severe accident instrumentation has already been completed for units 1 and 2, and will be completed in 2013 for unit 3 and in 2014 for unit 4 (HAEA 2012).

Accident management for Spent Fuel Pool

The Spent Fuel Pool (SFP) of each unit is located in the reactor hall, outside the containment. The SFPs have no second independent water supply or additional external water supply.

According to the valid emergency operating instructions, the water make-up (without electrical source) could be provided to the SFP by the gravity-forced discharge of water from the upper trays of the bubble condenser, while the lower trays are used for reactor cavity flooding. There is no confirmation so far that the amount of water in the bubble condenser would be sufficient for events affecting at the same time the reactor and the SFP. In order to improve safety, in the case of permanent loss of the ultimate heat sink (UHS), the licensee plans to implement a corrective measure assuring the long-term cooling of the SFPs by the establishment of a new, independent and protected supply route. According to the NAcP a new water supply route connected in the courtyard by flexible means shall be constructed until December 15, 2018. It shall be protected against external hazards. The spent fuel pool shall be filled from borated water reserve specified previously via this line. The required operations shall be specified in procedures (HAEA 2012).

For the SFP a PSA level 2 was performed. By ensuring an alternative water supply and alternative electric supply, a severe accident situation of the spent fuel storage pool might be managed. The consequence mitigating accident management to be executed subsequent to a severe accident of the spent fuel pool has not yet been prepared (HAEA 2011).

Severe accident management hardware provisions

As a result of the EU Stress Tests, the severe accident management hardware provisions shall be improved (HAEA 2011; HAEA 2012): Appropriately protected independent severe accident diesel generators shall be installed (final deadline December 15, 2018; by provision of appropriate electrical power supply it shall be established the bank filtered well plant be able to supply water to the essential service water system (final deadline December 15, 2015); for the construction of an external water supply route to the auxiliary emergency feedwater system, the necessary equipment shall be purchase (final deadline December 15, 2016).

One addition to the existing SAMGs is also planned: The method of usage of external supply opportunity shall be described in instruction documents until December 15, 2017 (HAEA 2012).

Management of multi-unit severe accidents

According to HAEA (2011), a basic principle applied to the Severe Accident Management Guidelines (SAMGs) is that every available system can be used during the management of the accident process. But the SAMGs refer to the alternative use of systems of the twin-unit, which is not possible in the case of a multi-unit accident. To remedy this situation, the resources and the accident electrical energy supply of the dedicated accident management system were installed individually on each unit; thus the management of severe accidents occurring on different units is made independently of each other, and the management of multi-unit accidents is solved from the technical point of view. On the other hand, the simultaneous accident management on more than one unit means increased organisational tasks that the personnel have to perform (see below).

The systems required for the simultaneous management of fuels stored in the spent fuel pool and in the reactor pressure vessel are available, but the guideline on the use of resources has not yet been prepared. The guidelines enter into force in the various units, when the respective technical modification are completed, regarding unit 1 and 2 in 2012, unit 3 in 2013 and unit 4 in 2014 (HAEA 2012).

Site organisation for accident management

According to HAEA (2011, 2012), the following measures are planned for strengthening the site organisation for accident management (sorted by date of final deadline):

- Because the plant is not fully prepared to manage liquid radioactive wastes generated in large quantities during a severe accident, procedures shall be developed for management of such large volume of contaminated water (final deadline December 15, 2015);
- Air-conditioning of the Protected Command Centre (PCC) shall be improved (final deadline December 15, 2015);
- Informatics mirror storage computers shall be installed both at the PCC and the Backup Command Centre (final deadline December 15, 2016);
- A nuclear emergency response centre resistant to earthquakes of a peak ground acceleration higher than design basis earthquake shall be established (final deadline December 15, 2016);
- A software-based severe accident simulator has to be established (final deadline December 15, 2017);
- The number of staff has to be determined; procedures have to be developed for personnel and equipment provisions (final deadline December 15, 2017);
- A Backup Command Centre equivalent with the PCC shall be established (final deadline December 15, 2017);
- For simultaneous management of severe accidents occurring on more than one (or even all) units, the physical arrangement and instrumentation at the PCC have to be extended (final deadline December 15, 2018);
- The radio communication has to be assessed in the case of permanent loss of electric power and earthquakes and the necessary actions shall be performed (final deadline December 15, 2018);
- A transportation vehicle providing adequate radiation protection under severe radiation conditions has to be purchased (final deadline December 15, 2018).

WENRA Reference levels

- After completion of the amendment of WENRA Reference levels the missing requirements will be incorporated in the nuclear safety requirements until December 15, 2018 (HAEA 2012).

Conclusion of the Stress Tests Peer Review

The Peer Review Team highlighted the following good practices as commendable (ENSREG 2012):

- The agreement between the utility and the regulatory authorities to update the PSA annually.
- The arrangements in place in the Protected Command Centre (PCC).
- The requirement of SAMG in the national regulatory framework.

The Peer Review Team concluded that the Hungarian approach to manage severe accidents seems to be comprehensive, major weak points for the severe accident management were not identified. Nevertheless, there are areas where further improvement may be achieved (ENSREG 2012):

- Full coverage of the issues associated with multi-unit accidents including severe damage to the infrastructure, and the issue of generation and distribution of hydrogen in the reactor hall during twin unit accidents.
- Upgrading the Back-up Commend Centre (BCC) against earthquakes, radiation, external temperature, etc.
- Suitable measures to prevent over-pressurisation of the containment have to be developed and implemented.

Furthermore, detailed studies on several topics are supported by the conclusions of the Peer Review Team (ENSREG 2012).

6.2 Evaluation and conclusions

Sufficient information about the structure of the containment, its design data, leakage rate and capacity under accident conditions[20] were given by the Hungarian side. It was reported that in the frame of the preparation procedure of the LTE licensing the overall review of the ageing management programmes relating to constructional components is in progress. Because of the importance of the confinement system for plant safety, more detailed comments would be of interest to the Austrian side (see chapter 7: Ageing Management).

The design leakage rate is 14.7 vol% per day for the maximum DBA case (LBLOCA). A full pressure test for unit 3 showed a lower leakage rate (4.8 vol% per day). The results of full pressure tests at the units 1, 2 and 4 have not yet been presented, but were of interest to the Austrian side.

Accidents with the steam generator (SG) tube or collector rupture lead to particularly high releases, since the containment is bypassed – unless effective countermeasures are implemented. Measures to be implemented at Paks NPP are briefly mentioned in a published article; due to time constraints, this topic could not be addressed in the Roadmap Process. This issue should be clarified in the frame of the above mentioned project “Stress Tests Follow-up Actions”.

The Hungarian side provided comprehensive information on Severe Accident Management (SAM) at Paks NPP in the course of the Roadmap LTE Paks NPP and by the national Stress Tests report. As a pre-condition for the LTE the nuclear authority required that the modifications necessary for the management of beyond design basis events and severe accidents shall be completed (HAEA 2011). The Severe Accident Management Guidelines (SAMGs) package was completed in May 2008. Before the usage of SAMGs is possible, however, technical modifications have to be implemented for some plant systems and equipment.

Depressurisation of the primary system is an important step of the SAM, in particular as a pre-condition for in-vessel retention of the molten core. At the 15[h] bilateral meeting in 2009, there was a discussion about requirements of valves for successful depressurisation. The issue could be regarded as closed.

[20] The containment behaviour under BDBA sequences is dealt with in chapter 6: Power Uprate and Fuel Development.

The installation process of hydrogen recombiners was accelerated after the Fukushima accident and 60 (30 pairs) passive autocatalytic recombiners are meanwhile installed at the containments of all units (HAEA 2011). According to the Stress Tests the possible hydrogen hazard in the reactor halls is an issue and information of this issue are of interest to the Austrian side.

One of the most important modifications is the external cooling of the reactor pressure vessel, which shall ensure its intactness during a severe accident. According to HAEA (2011), the calculations of the in-vessel retention (IVR) concept were justified in the frame of CERES experimental analyses. Because of its importance, it might be desirable that the issue of this IVR concept should be taken up again, in particular results from the CERES tests (see above). In any case, it will be of interest to the Austrian side to monitor the implementation of IVR at all four units at Paks.

During the slow increase of pressure caused by steam produced during the external cooling of the RPV, if means are not available to reduce the pressure, the unfiltered release through the stack will be necessary to avoid containment failure. Depending on the leakage rate, the containment pressure exceeds this value within 3 – 8 days. According to HAEA (2011), thus a system (filtered venting or containment internal cooling) aiming at the prevention of the slow overpressurisation of the containment has to be designed. According to HAEA (2012) Paks NPP prepared the installation of an active cooling system of the containment. Both, information of this system and the reason for the decision for this system, are of interest to the Austrian side.

The Peer Review Team concluded that the Hungarian approach to manage severe accidents seems to be comprehensive, major weak points for the severe accident management were not identified. Nevertheless, there are areas where further improvement may be achieved. Several improvements of the SAM, particularly regarding the management of accidents in the spent fuel pool and multi-unit accidents, are envisaged. Information about technical measures and studies would be of interest to the Austrian side.

According to HAEA (2011), the introduction of severe accident management significantly reduces the probability of large radioactive releases; it is expected that this value will not exceed the more strict requirements for newly built units. For the Austrian side information on SA source terms and large release frequencies after complete implementation of SAM strategy and mitigative actions is of high relevance.

6.3 Open questions/issues to be further addressed

Of the following questions/issues, all but the first one will be treated in the framework of the above mentioned Project "Stress Tests Follow-up Actions", topic 3. The first issue could be discussed, as appropriate, at the regular bilateral meetings.

- The design leakage rate is 14.7 vol % per day for the maximum DBA case (LBLOCA). A full pressure test for unit 3 showed a lower leakage rate (4.8 vol% per day). The results of full pressure tests at the units 1, 2 and 4 would still be of interest to the Austrian side.

- Accidents with the steam generator (SG) tube or collector rupture lead to particularly high releases, since the containment is bypassed. Measures are planned or already implemented at Paks; due to time constraints, this topic could not be addressed. The clarification of this issue would still be welcome.
- Because of its importance, the in-vessel retention (IVR) concept should be taken up again in case the Austrian experts identify remaining open questions concerning the results from the CERES tests. The implementation of IVR concept at all four units in Paks should be monitored in any case.
- For the Austrian side information on SA source terms and large release frequencies after implementation of SAM strategy and mitigative actions is of high relevance.
- As a result of the Stress Tests several improvements of SAM, particularly regarding the management of accidents in the spent fuel pools and multi-unit accidents, are envisaged. Information about the planned measures and results of studies would be of great interest to the Austrian side, especially regarding:
  - the active containment cooling system aiming at the prevention of the slow over-pressurisation of the containment,
  - the water supply to the spent fuel pool from an external source.

7 AGEING MANAGEMENT

Ageing means changes of design characteristics during operation. All systems, structures and components (SSCs) are subject to ageing due to specific mechanisms. The relevance of ageing for safety is reflected by the objectives of regulatory supervision of ageing management as stated in the HAEA regulatory guideline 1.26:

- Maintenance of integrity of defence-in-depth boundaries;
- Prevention of increasing of system component failure probability;
- Maintenance of system and equipment performance indicators;
- Protection against common cause failures.

Therefore, a comprehensive system of ageing management is required for an NPP, particularly in case of life time 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.

7.1 Summary of information provided

In the Preliminary Environmental Study (PES 2004) it was stated that an ageing management programme was implemented at Paks NPP, which is being developed further as part of the planning process for the lifetime extension. According to PES (2004) systematic monitoring of ageing was begun several years ago, focusing on the reactor pressure vessel embrittlement, and erosion corrosion.

Furthermore, a programme of registration of ageing effects, description of the changes, and determination of corrective action was mentioned. The results of the programme concerning ageing effects, including brief indications which measures are required in case of a lifetime extension to 50 years, were listed in PES (2004). However the ageing management programme was not described in detail. Furthermore, the listing was restricted to building structures and mechanical components and systems (including emergency diesel generators, ventilation, off-gas treatment and waste water treatment). The complex of electrical and I&C-systems was summarily dealt with.

In the answers of Paks NPP to the study of Umweltbundesamt (ANSWERS 2006) more information on the ageing management programme was provided. It was 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 programme for safety-related passive components had to be reviewed by requirement of the licensing body. This review was performed according to the methods applied by U.S.NRC in the course of license renewal, considering ten main steps. Ageing management of the large number of active components is being monitored by the maintenance effectiveness monitoring system, which was introduced at that time.

On the basis of this information, the Austrian experts judged that the ageing management programme was not fully completed at that time, particularly not in the context of lifetime extension. Parts of the system appeared to be well implemented and, for some years, successfully performing. It was pointed out that the regulatory system in Hungary was in the process of being changed, regarding in-service-inspections, which constitute the basis for ageing management. The respective plans included plans by the licensee to reduce the frequency of in-service-inspections for safety-relevant equipment. In this context new approaches were under development, e.g.:

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

Those new trends and approaches should be followed, including the evolution of IAEA activities (IAEA was preparing a Safety Guide "Ageing Management for Nuclear Power Plants and Research Reactors" at that time).

Additional information was provided by the Hungarian side during the discussion at a public hearing on June 6, 2006 where 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 summarised. It was reported that in the context of the lifetime extension, the current ageing management programme is being reassessed, applying 10 criteria as required and defined by U.S.-regulations. This re-assessment was almost concluded at that time; it mostly led to a confirmation of the existing system, with only a small number of modifications required. Concerning the DACAAM, it was reported that this system, developed in Hungary, has recently been acquired by the operators of Loviisa NPP. All the information was provided orally at the above mentioned hearing.

Based on the information available up to June 2006 it was concluded in a report to the Austrian Government (UMWELTBUNDESAMT 2006) that considerable changes and developments were to be expected in the ageing management programme of Paks NPP during the next years. Further information on this process was judged to be of great interest from the Austrian point of view. In particular, further observation of this issue should permit to ascertain that the new approach to in-service-inspections to be introduced at Paks NPP, which is to include reductions in inspection efforts without a decrease of the safety level, indeed does not lead to any safety level decreases.

In advance of the 15th bilateral meeting the paper "Development of ageing management and in-service inspection (concerning steam generators); steam generator corrosion" by János Pinczés (Paks NPP) (PINCZES 2009) was submitted to the Austrian side. It contains information about Hungarian regulatory guidelines for ageing management, their implementation in Paks NPP, in-service inspection of steam generators and experiences with steam generator corrosion.

The paper PINCZES (2009) contains an overview of the relevant regulation in Hungary including a table showing the correspondence between regulation concerning ageing management and in-service inspection (see table[21] 7-1). The Hungarian regulation was established according to the advanced international ageing management practice (ageing management regulation in USA, IAEA guidelines).

The information concerning plant specific ageing management, which is contained in PINCZES (2009), is structured according to the following aspects:

- ageing management programme,
- component specific ageing programmes,
- ageing mechanism driven ageing programmes,
- data collection and analysis for ageing management,
- component specific ageing programme for the steam generator,
- in-service inspection,
- steam generator corrosion.

Ageing management programme

It is stated in PINCZES (2009) that the possibility of the life time extension is strongly linked to the ageing management of safety related components (Safety Classes 1-3+).

At Paks NPP, the ageing management activities are performed according to a "Comprehensive ageing management procedure" and additional procedures for component specific ageing management programme. Paks NPP performed a review of its ageing management programme (AMP) on basis of the document NUREG-1801 "Generic Ageing Lessons Learned (GALL) Report".[22] On the basis of the results of this review some modifications and additional one time inspections of safety related equipment were implemented.

It is stated in PINCZES (2009) that the addition of these procedures in 2009 provided full compliance with the NSC and the Safety Guidelines 4.12. and 1.26. Completion of the ageing management programmes was planned for end of 2009. According to SG 4.12, the comprehensive AMP has to contain a recognition of degradation mechanisms occurring in the plant (type programmes), a

[21] All figures and tables are presented in Annex 1.

[22] NUREG-1801 is referenced as a technical basis document in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR). The GALL Report identifies ageing management programmes (AMP), which were determined to be acceptable programs to manage the ageing effects of systems, structures and components (SSC) in the scope of license renewal, as required by 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

specification of the scope of ageing management, an ageing review of components belonging to the scope (component/component group specific AMPs), a modification of existing and elaboration of new programmes based on the review. The coordinated operation of the programmes and reviews is required.

The process of ageing management at Paks NPP concerns the activities of several departments of the plant (mainly the departments for maintenance, ISI, technical engineering, operation and the safety directorate). Ageing management activities are mainly governed by operational, maintenance and inspection procedures (ISI, In-service test, technical supervision etc.). A separate ageing management section has been established at Paks NPP for harmonisation and coordination between the technical support organisations and the in house technical departments.

According to SG 4.12, the equipment is to be divided into highlighted system components and system component commodity groups. Highlighted system components are those with prominent safety role or special ageing management importance. An individual AMP has to be prepared and operated for these components, i.e. they are managed separately. System component commodity groups are made of non-highlighted components based on their identical features (e.g. base material, operating medium, operating parameters, constructional design). Primary aspect for definition of the groups is similar ageing. A minimum scope of highlighted system components is specified in Annex 1 "Guideline for the specification of the scope of the comprehensive ageing management program" of SG 1.26. It contains the following components: reactor pressure vessel and its internals, reactor control rod driving mechanism, reactor supporting structures, pressuriser, steam generators, main gate valves, main circulating pumps, main circulating loops and seismic reinforcements of major components.

The AMP of the highlighted and separately managed SSCs is task of the ageing management section. Other safety relevant SSCs are managed by equipment engineers with the ageing management section supporting this work. Ageing management of civil structures (buildings) and I&C equipment is performed by the responsible organisations.

Component specific ageing programmes

Based on the requirements in SG 4.12, component specific and component commodity group specific AMPs were developed for highlighted SSCs and for SSCs within the commodity groups. Periodic reviews of the performance of the component specific programmes allow for modifications of existing programmes or for the definition of new programmes. The preparation, publishing and possible modification of component specific AMPs is part of the work of the ageing management Section.

It is stated in PINCZES (2009) that in 2009 identical component specific ageing programmes and appropriate ISI programmes were already established for the highlighted components while for the other SSCs the final implementation of the ageing programmes was under way. The content of each component specific AMP complies with the 10 issues stated in the Safety Guideline 4.12 (SG 4.12):

1. degradation processes, determination of ageing sensitive structural locations;

2. introduction of actions mitigating or preventing ageing processes;
3. designation of parameters to be monitored;
4. detection of ageing effects;
5. monitoring, trending;
6. specification of acceptance criteria;
7. introduction of corrective actions;
8. feedback, improving efficiency of AMP;
9. administrative control;
10. utilisation of operating experience.

The component specific AMPs contain lists of the parameters to be observed in the context of trend analysis. The purpose of trend analysis is to detect changes in ageing tendencies at an early stage. Requirements for in-time detection, prevention and mitigation of ageing mechanisms are fixed in respective procedures. The organisation, which performs a certain inspection or testing programme, generally is responsible for detection and monitoring of ageing effects, first steps of the trend analysis, and equipment condition evaluation.

The component specific programmes also contain the acceptance criteria or respective references for monitored parameters. The acceptance criteria are based on analyses and calculations. In case a certain parameter is adverse than the acceptance level, the respective component will be repaired or changed. In specific cases other measures could take place (preventive or mitigating actions to stop or slow down ageing mechanisms). Especially the following failure modes should be avoided by suitable monitoring:

- stable and unstable crack propagation at the boundary of pressure components and pipelines,
- opening of leakage routes causing failure,
- erosion effect of leaking operating medium,
- acceleration of local corrosion processes (stress corrosion etc.),
- exceeding of environmental load parameters considered in equipment qualification.

For the highlighted components, the ageing inspections are performed in frame of the ISI programmes. The methods to be applied to ISI and the period between inspections are based on design requirements and analyses. In general 100% inspection has to be performed within a cycle of 4 years based on the requirements of the designer (Russian rules).[23] After adaptation of ASME Boiler and Pressure Vessel Code (BPVC) sections to Paks NPP the period should be doubled to 8 years.

Ageing mechanism driven ageing programmes

Among other things, the ageing management section at Paks NPP is also responsible for the collection of the available knowledge about the ageing mechanisms in the context of an ageing mechanism driven ageing programme. Purpose of this programme is to collect and evaluate information about the relevant ageing mechanisms. It is pointed out in PINCZES (2009) that Paks NPP established

[23] For the SG heat exchanger tubes the inspection cycle for 100% is 12 years according to PINCZES (2009).

programmes for all ageing mechanisms that were detected at the plant or could possibly be expected according to the international practice and knowledge. Most of the ageing mechanisms are connected to corrosion. In 2009, altogether 19 ageing mechanism driven ageing programmes[24] were implemented at Paks NPP.

Data collection and analysis for ageing management

According to PINCZES (2009) the database/expert system DACAAM (Data Collection and Analysis for Ageing Management) was implemented in Paks NPP. In the frame of DACAAM those data and documents are collected, which are recommended to be recorded and frequently assessed. At least the following SCCs are covered by the DACAAM system: reactor pressure vessels, reactor internals, steam generators, main circulating piping, pressurisers, surge pipelines, main circulating pumps, main gate valves, main feed water piping, main steam piping, other safety related pumps and valves, safety related heat exchangers and containment penetrations.

The structure of the DACAAM system mainly follows IAEA recommendations for an appropriate PDCA cycle for ageing management (PDCA: plan-do-check-act). The following relevant data are contained in the DACAAM System:

- Regulatory requirements.
- Baseline information. Among others baseline information contain construction data (e.g. dimensions, materials, "0" condition defects/deficiencies data); design information (e.g. expected neutron flux, forecast for evolution of the toughness of irradiated materials, stress calculation results); design specifications; degradation mechanism forecasting information; component identification (including component type and location); expected degradation mechanisms and potential critical sites descriptions; locations susceptible to local corrosion mechanisms; data of component installation and design modification data.
- Operation history data. Each component has been exposed to specific conditions which ought to be contained in the operation history data, including data on process conditions, chemistry, transients as well as testing and failure data. Typical data are: pressure; temperature; flow rates; neutron flux; water chemistry data (e.g. pH, concentrations of impurities); material surveillance data; operational cycle counting data. It is pointed out in PINCZES (2009) that the knowledge of the operating history data is essential for an effective ageing management: they are the prerequisite for an evaluation of a component's design life usage (e.g. concerning fatigue[25]) and they allow the differ-

[24] General corrosion, Boric acid corrosion, Erosion-corrosion, Stress corrosion cracking , Local corrosion (for example pitting) , Microbiological corrosion, Irradiation assisted stress corrosion, Erosion of the ground, Swelling, Erosion, Temperature stratification, Low-cycle fatigue, High-cycle fatigue, Wear, Loosening, Deposit, Embrittlement due to irradiation, Thermal ageing, Water hammer.

[25] „For the primary system pressure boundary components of a PWR, design rules require a fatigue assessment based on a list of transients which are supposed to represent the entire life of the plant. Of course, this assessment is meaningful only if during operation plant staff verifies that all actual transients are not more severe or more numerous than assumed in the design analysis. When it is done properly, transient monitoring and documentation give, at any time, a clear view of where each component stands with respect to its fatigue margins." (PINCZES 2009)

entiation of ageing related failures from other failures and the identification of specific environments favouring degradation.

- Maintenance history data. Typical maintenance history data are: component condition indicator data (e.g. results of in-service inspections); date, type and description of the maintenance/ISI programme; degradation failure management description. As is pointed out in PINCZES (2009) also routine information such as test results or monitoring data can provide useful insights. Even in cases where test results are in compliance with the technical specification, the data may be valuable for trend analysis. As these data are usually collected by the production personnel and evaluated by engineering personnel clear instructions have to be provided to allow adequate processing of the data. These concerns the interfaces between the AMP and the plant's organisations/sections responsible for data collection.
- Ageing management programme experience data. Typical data are: degradation mechanism forecasting data; degradation mechanism root cause analysis data; domestic and international ageing related events data; construction materials/environment/degradation occurrences data trending.

Component specific ageing programme for the steam generator

Steam generators (SG) are expected to be susceptible to the ageing mechanisms fatigue, general corrosion, boric acid corrosion, local corrosion, wear, loosening, thermal ageing, deposit. A list of 10 locations sensitive for ageing effects is presented in PINCZES (2009). They have been determined on basis of the annex of the Regulatory Guideline 1.26 and the ageing mechanism driven ageing programmes. For all of these sensitive locations the AMP refers to instructions (maintenance or operation) or ISI programmes for detection and evaluation of the relevant ageing effects. In case preventive or mitigating actions are possible respective information is also contained in the programme.

The ageing programme for the SG includes a description of operation history data concerning detected ageing mechanism effects. Trends important for the SG lifetime are evaluated. Especially for SG heat exchanger tube plugging the reserve of the heat transfer surface at the end of the long term operation is predicted. It was possible to positively influence this trend (reduced number of plugged tubes per period) with the aid of some modifications of the secondary equipment and changes in the secondary water chemistry parameters (high pH value).

Quality of the ISI methods is also mentioned in PINCZES (2009). Eddy current ISI of heat exchanger tubes was qualified according to methodology of the European Network for Inspection Qualification (ENIQ). In 2009, qualification of the ultrasonic ISI of the NA 500 primary nozzles (transition weld) was going on. According to PINCZES (2009) the inspection of this dissimilar weld is very complicated. Furthermore the availability of qualified inspection methods is important for the application ASME BPVC code sections.

The ageing management programme for the SGs comprises 100 pages. Main aspects of the programme are explicated in PINCZES (2009) for the ageing mechanism "general corrosion". Locations expected to be sensitive for general corrosion are SG shell/welds/nozzles, flange joints and seismic reinforcement and directly connected holders. Preventive or mitigating actions are painting with anticorrosion paint and the repair of the damaged painted surface (PINCZES 2009):

> *"The program precisely determines during the maintenance work which program has to be used for repair or new painting. Controlled parameters are: surface changes and/or loss of the wall thickness. The program determines organisations performing the control, the steps of the document preparing and requirements for establishing the Technical Review Plan. Chapter: detecting ageing effects determines the maximum period between ISI and listing all the documents (with exact pages and section number) with the inspection information and requirements. The section monitoring, trend analysis and condition monitoring information is that in case of the small defects, repair works trend analysis and monitoring not required. Evaluation is performed by Ageing Section in case these are the inspection and maintenance results. The results of the evaluation are presented in the ageing management annual report. Acceptance criteria are: the minimum acceptable wall thickness. Value of the minimum acceptable wall thickness included in the Acceptance Criteria or in case will be calculated. For the general corrosion the Corrective action is: repair the damaged painting or use an advanced paint, painting method. The program describes the painting repair or change process. Next chapter of the program is not divided according to the ageing mechanisms. Review of the SG ageing information has to be presented in the annual ageing report."*

The DACAAM screen for SG heat exchanger tubes is shown in figure[26] 7-1. It can be seen, the relevant data can be made accessible by point-and-click.

The visualisation of ISI and maintenance history data for SG heat exchanger tubes is shown in PINCZES (2009) as an example. Indications and information about the plugged tubes are displayed with a special 3D data visualisation tool (figure[27] 7-2). According to PINCZES (2009) more than 80% of the indications in the horizontal steam generator tube bundles are at the position of tube supports plates, where secondary side corrosion products with concentrated corrosive agents preferentially accumulate.

The documentation of age related failures is carried out by the personnel responsible for maintenance, ISI or by other specialists. The respective documents are stored in or linked to the DACAAM system in a special format, enabling AMP related trend analysis and/or event reporting.

In-service inspection (ISI)

The inspection of the highlighted separately managed SSCs is performed according to component specific ISI programmes which have to be licensed by the Authority. The relevant criteria for the evaluation of the results of all types of inspection methods applied within the ISI programmes are collected in a document which has to be licensed by the Authority too. A document with revised acceptance criteria suitable for the application of ASME BPVC has been sent to the Authority for the license.

[26] All figures and tables are presented in Annex 1.

[27] All figures and tables are presented in Annex 1.

According to PINCZES (2009), the following aspects are part of the ISI programmes:

- validity information,
- reference documents,
- necessary technical conditions to perform the inspections,
- other conditions important to perform the inspection,
- safety and radiation safety rules,
- subject of the inspection,
- inspection methods and evaluation criteria,
- inspection documentation,
- as a notice the inspections which have to be performed after sealing and information about the percent of the performed inspections if the volume is not 100%,
- inspection table,
- necessary drawings.

As an example a part of the inspection table of the SG ISI programme is shown in PINCZES (2009). It has already been modified for the ASME BPVC Section XI adaptation. It contains information concerning main parts and elements of the equipment, inspection method/technology, inspection category according to ASME, qualification of the inspection, inspection volume and inspection cycle.

Concerning the procedures for the evaluation of test results those for the SG heat exchanger tubes (including adaptation of the scope of testing to the test result) are provided in PINCZES (2009).

Steam generator corrosion

An outline of activities for the management of steam generator corrosion is presented in PINCZES (2009). Generally three main types of corrosion processes are to be expected:

- corrosion without stress (general, local, and selective),
- stress corrosion (stress corrosion cracks, corrosion fatigue),
- corrosion due to service medium flow (flow accelerated corrosion or erosion corrosion).

Their relevance depends on the location sensitive for ageing effects. The respective relevance of the different processes for each of the 10 locations is shown in tabular form. Main characteristics of the corrosion processes are described.

Water chemistry is an important factor with regard to the corrosion processes in the SG. It also depends on equipment construction materials. A large number of hardware modifications were implemented in Paks NPP to remove copper from the feedwater-steam-circuit. Materials containing copper were replaced by stainless steel. Also main parts of the deposits were removed from the SG secondary side. Extensive investigations concerning the influence of water chemistry on SG corrosion processes were performed. So an adequate basis for the respective ageing management of the SG has been established. It is expected by Paks NPP that operation with the optimised secondary side water chemistry parameters will significantly improve the situation concerning growing of cracks or initialisation of the local corrosion processes. Therefore the control and evaluation of the water chemistry parameters is an important part of the ageing management activities. This also concerns the control and evaluation of the water chemistry in case of abnormal operation conditions (e.g. turbine condenser leakages).

Concerning corrosion processes with crack formation cracks were observed only in threaded holes of the primary collector flange area and on the secondary surface of the heat exchanger tubes. Therefore the collector head was changed. SG heat exchanger tubes with defect deepness of 50% of the wall thickness have been and will be plugged. In certain cases tubes can be left unplugged, then the propagation of the defect is controlled every year.

After each outage the trend concerning plugging of heat exchanger tubes is reevaluated for the SG which was inspected. According to PINCZES (2009) the trend is significantly below the maximum appropriate level (with respect to the retention of a sufficient heat exchange area at the end of the lifetime) even for the SG with the largest number of plugged tubes (approx. 2.6%).

On the basis of the requirements in the Hungarian Regulatory guidelines on ageing management and the information contained in PINCZES (2009) several questions were prepared for the 15th bilateral meeting. However not all of them could be discussed at the meeting, because of time constraints. Some questions also seemed to be too detailed for the discussion in this meeting with participants of very different background. Questions concerning the following aspects of the AMP at Paks NPP were discussed:

- Safety factors: According to chapter 6 of the HAEA guideline 3.13 “Consideration of ageing process during design” there is the possibility that the safety factors of components may be specified as a function of the lifetime. During the meeting it was declared that a reduction of safety factors is not admissible in NPP Paks.
- Application of ASME BPVC Section XI (“Inservice Inspection of Nuclear Power Plant Components”) to components of Paks NPP: It was stated during discussion that VVERs were not designed and built according to the ASME rules. Although the ASME boiler and pressure vessel code is a complete system (in general, there is a strong connection between material properties, design, fabrication, acceptance criteria and testing) it seems worthwhile for Paks NPP to use at least parts of the code if US materials could be connected to RU materials. Additionally ASME Code section III (“Rules for Construction of Nuclear Power Plant Components”) was also applied to Paks NPP components. It was stated by Paks NPP that too frequent ISI could be detrimental for the equipment, if components have to be dismantled too often. Hence, 4 years cycles are very short. It can be demonstrated that 8 years cycles are acceptable, since defects cannot grow is limited during this time. Application of ASME-Code sections to components of Paks NPP had not been decided/admitted by the Authority in 2009.
- Completeness of operating history data: According to Paks NPP a complete operating history data is available for the highlighted separately managed components. It is part of the data base.
- Inspection of SG and SG heat exchanger tubes: According to Paks NPP 80% defect deepness of the wall thickness would be acceptable as criterion, but this value would be highly unusual internationally. Hence, to avoid discus-

sions a value of 50% has been chosen. The integrity of the tubes is guaranteed for all transients in case the wall thickness is reduced to 50%. At the time of the meeting there very little plugging with improved trend because of changes in the water chemistry.

Additional information on AMP implementation at Paks NPP and especially on the modification of the ISI programme is contained in open literature. However a systematic evaluation has been beyond the scope of this report. Among the available publications only (TRAMPUS 2010) should be mentioned here as this text further explains the motivation for the application of the ASME BPVC section XI. According to (TRAMPUS 2010) the current ISI programme in Hungary is based on the former Soviet normative technical documents representing the technical level of the 1960s and 1970s. It is stated that even the later revisions, published in 1989, do not adequately follow the technical developments necessary for assessing structural integrity of components. Additionally the basic requirements in the older normative technical documents for design, manufacturing, commissioning and operation do not handle the evaluation of pressure-retaining components and piping during operation, i.e. they do not support plant life time extension. As a result, Paks NPP is aiming to adopt the ASME BPVC Section XI requirements for activities related to the safety of pressure retaining components according to (TRAMPUS 2010):

"In particular, these activities would include ISI, repair and replacement in case of inadequate ISI results, and strength and fracture mechanics analyses. The Hungarian nuclear safety regulation allows for their adoption, since it does not determine exclusively the codes or standards that must be used during the design and commissioning of a nuclear power plant, or for the ISI to be performed during operation. Instead, its requirements for the ISI programme only prescribe that they shall be specified in accordance with 'authoritative technical standards'.

The main goals of the adoption of BPVC Section XI are the improvement of the safety and of the cost-effectiveness of plant operation and maintenance. BPVC Section XI's operation and maintenance technical support will make state-of-the-art implementation possible. It will also enable inspection, maintenance and necessary safety analyses to be compared directly with the most advanced safety requirements and methods. The change has an indirect goal as well, namely to facilitate the international acceptance of the operational life extension plans of Paks NPP. Compliance with the BPVC Section XI requirements provides the opportunity to extend of the current four-year ISI cycle for class 1 components to an eight-year one for the whole operational life of the plant. The intended doubling of the inspection interval has a substantial potential to enhance the cost-effectiveness of future operation and maintenance."

7.2 Evaluation and conclusions

In Hungary, several regulatory guidelines for AMP and ISI have been implemented. These requirements define the basic scope of the AMP at Paks NPP. Based on the time frames mentioned in PINCZES (2009) meanwhile full implementation of the AMP should have been accomplished. A description of certain aspects concerning the AMP has been provided in PINCZES (2009).

Based on the available information we conclude that a comprehensive and systematic approach for ageing management has been implemented in Paks NPP – at least this applies to mechanical components, as no information concerning ageing management of structures and I&C components is contained in PINCZES (2009). The database/expert system DACAAM is an integral part of the AMP. Based on the available information it seems to be well suited for this purpose.

With the exception of activities concerning steam generator corrosion no detailed information is available with respect to the experience concerning the performance of the AMP in Paks NPP. Aspects concerning these experiences are e.g. the efficiency of the co-ordination and cooperation of the different departments responsible for certain aspects of ageing management as well as recent leakage events. According to the HAEA's report on "Recent Developments in Nuclear Safety in Hungary" from April 2011 (HAEA 2011a), leaks of a steam generator drainage pipe and a water purification system pipe in unit 4 were observed. According to HAEA the material testing results showed that the water purification system pipe failure was presumably caused by thermal fatigue. Because of previous similar failures an extensive investigation is in progress on the purification system pipes and the sampling pipe weldings. In HAEA's report it is not mentioned whether these events could have any consequences for the AMP in Paks NPP.

According to the HAEA's report on "Recent Developments in Nuclear Safety in Hungary" from November 2012 (HAEA 2012a) the Hungarian regulation's licensing procedure of the extended period shows similarity to the U.S. NRC approach in license renewal according to 10 CFR 54. The new Hungarian regulatory rules do not explicitly determine the applicable codes and standards neither for plant construction nor for ISI and in-service-testing. Therefore, adoption of ASME code sections is admissible in principle. The most fundamental objectives of ASME adoption are the review and adjustment of the plant's ISI and in-service-testing programmes to meet the ASME Code XI. requirement. From HAEA's perspective, this needs careful consideration as Paks NPP has not been constructed, commissioned and operated up to now in line with the relevant sections of ASME Code. According to HAEA's report the task is being implemented. The Hungarian Standardization Institution plans to issue ASME III. and XI. Code edition as a Hungarian Standard (MSZ 27003 and 27011) in Hungarian language.

The adoption of ASME Code section XI necessitates a post evaluation of the materials, the design and the operation of the relevant components. Up to now no detailed information about the approach for ASME code adoption and the doubling of ISI cycle length has been presented.

7.3 Open questions/issues to be further addressed

Based on the information summarised in chapter 7.2 and on the evaluation in chapter 7.3 we recommend that the following issues should be further addressed as appropriate in the bilateral process between Hungary and Austria in the framework of regular meetings:

- The ageing management programme for structures and I&C components.

- The experiences with respect to the general performance of the ageing management programme in Paks NPP. Aspects concerning these experiences are e.g. the efficiency of the co-ordination and cooperation between the different departments responsible for certain aspects of ageing management as well as recent leakage events (steam generator drainage pipe and a water purification system pipe in unit 4).
- The adoption of ASME Code section XI. Aspects concerning this adoption are a post evaluation of materials, design and operation of the relevant components.
- The technical justification for the doubling of in-service inspection (ISI) cycle length.

8 TERROR ATTACK

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.

8.1 Summary of information provided

Neither in the Preliminary Environmental Study nor in the Environmental Impact Study, malicious acts of third parties against Paks NPP and their possible effects are discussed (PES 2004, EIS 2006).

It is stated in the (EIS 2006), that external impacts like airplane crash and explosions are regarded to be very unlikely and hence are not considered either. Regarding the design of the reactor building, it is stated that the upper part is built like any ordinary industrial building (EIS 2006). Furthermore, it is stated that the probability of the crash of an aircraft onto the plant is so small that this event need not be considered, which also indicates that design against crash of an aircraft was not regarded as necessary. According to information provided at the 14th bilateral meeting in 2008, the wall thickness of the concrete structure of containment is between 0.5 m–2 m.

If an accident occurs in the spent fuel pool, radioactivity would be released directly to the reactor hall and from there to the environment. As a result, the effects of the release could be significant, although the environmental consequences would be less severe than for a beyond design basis or severe reactor accident due to the decay period of the fuel (HAEA 2011). An accident in the spent fuel pool could be caused by a terror attack.

The Answers provided by Paks NPP emphasise that the NPP meets the legal requirements concerning physical protection (ANSWERS 2006). 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[28] 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 September 11, 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 organisations 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.

According to the agreed time schedule of the LTE Paks Roadmap terror attack were discussed at the 15th bilateral meeting. A reference paper for this topic was not provided by the Hungarian side. At the meeting, there was a presentation “Nuclear Security in Hungary” by Árpád Vincze (referent) and Kristóf Horváth (HAEA).

The presentation provided general information on nuclear security and non-proliferation, as well as on physical protection (legal basis, outline of approach). For Paks NPP, it was stated that the threat and vulnerability analysis was revised in 2007–2008. A new, elevated design basis threat (DBT) was defined,

[28] 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

providing additional capabilities to defend against incoming threats. Further analyses in 2008–2009 showed that current physical protection requirements are met; however, scope for improvements was found.

No details concerning the new DBT were provided. In reply to a question of the Austrian side, the Hungarian Authority emphasised that IAEA guidelines were followed, and not the new approach in the US (where the DBT was recently revised by NRC). It was pointed out that threat perception in Europe and in the USA differs considerably.

In 2011, a modification of the Atomic Act takes place in Hungary: The ten IAEA Basic Safety Principles are now included in the legislation. Issues related to nuclear security are also included in the Act. HAEA is working on a definition of the Design Basis Threat (DBT)[29].

8.2 Evaluation and conclusion

The information provided at the 15th bilateral meeting gave only very general information which is not sufficient by far to disprove that large radioactive releases are possible after a terror attack. Indeed, these hazards exist for all commercial nuclear power plants; in addition, there seem to be some specific vulnerabilities at VVER-440/213 plants.

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.

It must be emphasised 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.

The European Stress Tests were conducted along two parallel tracks: Safety Track and Security Track. It is the aim of the Security Track to analyse security threats and a methodology for the prevention of, and response to, incidents due to malevolent or terrorist acts. For the assessments under this second track, the Council set up the Ad-hoc Group on Nuclear Security (AHGNS) (EC 2012).

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

[29] Presented information is taken from HAEA’s short reports on “Recent Developments in Nuclear Safety in Hungary”, issued in April and October 2011

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.

8.3 Open questions/issues to be further addressed

Further information regarding the issue of terror attacks (e.g. DBT) would be of great interest to the Austrian side, considering the large consequences of potential attack. Vulnerabilities, attack scenarios and potential consequences can and should be discussed in an appropriate general manner. Due to the sensitivity of the topic, discussion would require an appropriate framework.

.

9 REFERENCES

ANSWERS (2006): Answers to the study of Umweltbundesamt, to the statements of the provincial authorities of Lower Austria, Burgenland and Wien, Greenpeace, Global 2000 etc. provided within the framework of the EIA process for the lifetime extension of Paks NPP by the Hungarian side.

BAREITH, A. et al.(2003): Seismic PSA for NPP Paks of Hungary; In: Transactions of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17, Paper # M02-5); Prague, Czech Republic, August 17–22, 2003.

BAREITH, A. (2007): Use of Insights from Seismic PSA for NPP Paks; in: Specialist Meeting on the Seismic Probabilistic Safety Assessment of Nuclear Facilities NEA/CSNI/R(2007)14; Jeju Island, Republic of Korea; 2006.

BFS – Bundesamt für Strahlenschutz (2005): Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke; Stand: August 2005.

BUS, Z.; GRENERCZY, G.; TÓTH, L. & MÓNUS, P. (2009): Active crustal Deformation in two seismogenic zones of the Pannonian region – GPS versus seismological observations, Tectonophysics, 474 (2009) p. 343–352.

CONTAINMENT (2008): Containment behaviour during DBA and BDBA events at Paks NPP, Provided for the 14th bilateral meeting in 2008 by the Hungarian side.

EC – European Commission (2012): Technical summary on the implementation of comprehensive risk and safety assessments of nuclear power plants in the European Union; Commission Staff working document; Brussels, October 4, 2012.

EIS (2006): Environmental Impact Study for LTE of Paks NPP.

ELTER, J.; TÓTH, E.; LAJTHA, G. & TÉCHY, Z. (2009): Development of the SAM strategy for Paks NPP on the basis of Level 2 PSA; Workshop on Implementation of Severe Accident Management Measures (ISAMM), Schloss Böttstein, Switzerland, October 26–28, 2009.

ELTER, J. (Paks NPP) (2010): Insights of the seismic risk assessment and seismic upgrades, Paks nuclear power plant. Presentation handed to the Austrian delegation at the 16th Bilateral Meeting between Hungary and Austria, 22/23 November, 2010. Schloss Hernstein, Austria.

ENSREG – European Nuclear Safety Regulator Group (2012): Stress Tests performed on European nuclear power plants; Peer review country report for Hungary, April 2012; http://www.ensreg.eu/EU-Stress-Tests/Country-Specific-Reports/EU-Member-States/Hungary.

EPRI – Electric Power Research Institute (2006): Use of CAV in Determining Effects of Small Magnitude Earthquakes on Seismic Hazard Analyses; DRAFT Final Report, June 2006.

GODOY, A.R. & GÚRPINAR, A (2001): Technical guidelines for the seismic safety re-evaluation at Eastern European NPPs; in: Seismic evaluation of existing nuclear facilities; IAEA-TECDOC-1202, March 2001.

GÚRPINAR, A & ZOLA; M. (2001): Results of a benchmark study for the seismic analysis and testing of WWER type NPPs: Overview and general comparison for PAKS NPP; in: Seismic evaluation of existing nuclear facilities; IAEA-TECDOC-1202, March 2001.

HAEA – Hungarian Atomic Energy Authority (2011): National Report of Hungary on the Targeted Safety Re-assessment of Paks Nuclear Power Plant, December 29, 2011; **http://www.ensreg.eu/EU-Stress-Tests/Country-Specific-Reports/EU-Member-States/Hungary**.

HAEA – Hungarian Atomic Energy Authority (2011a): Recent Developments in Nuclear Safety in Hungary, April, 2011; **http://www.haea.gov.hu/web/v2/portal.nsf/download_en/42A7926CF3D243BFC1257881002EC1B5/$file/Recent%20Developments%202011_1.pdf**

HAEA – Hungarian Atomic Energy Authority (2012): National Action Plan of Hungary on the implementation actions decided upon the lessons learned from the Fukushima Daiichi accident, December 19, 2012; **http://www.ensreg.eu/sites/default/files/Hungary%20NAcPv4_Eng_final_small_0.pdf**

HAEA – Hungarian Atomic Energy Authority (2012a): Recent Developments in Nuclear Safety in Hungary, November, 2012; http://www.haea.gov.hu/web/v2/portal.nsf/att_files/brochur/$File/RecentDevelopments2012_2.pdf?OpenElement

HIRSCH, H. & BECKER, O. (2005): Vulnerability of Nuclear Power Plants to Terror Attacks: An Overview; Enersec Conference Series, Proceedings of Conference "Nuclear Energy and Security", Salzburg, July 20–23.

HORVATH, F. & BADA, G. (2004): Atlas of the present-day geodynamics of the Pannonian basin: Euroconform maps with explanatory text; 2004. **http://geophysics.elte.hu/projektek/geodinamikai_atlasz_eng.htm**

IAEA – International Atomic Energy Agency (1993): Probabilistic safety assessment for seismic events; IAEA-TECDOC-724, October 1993.

IAEA – International Atomic Energy Agency (2003a): Seismic Evaluation of Existing Nuclear Power Plants; Safety Reports Series No. 28; Vienna.

IAEA – International Atomic Energy Agency (2003b): Site Evaluation for Nuclear Installations. Safety Requirements, NS-R-3; Vienna.

IAEA – International Atomic Energy Agency (2009): Evaluation of Seismic Safety for Existing Nuclear Installations; Safety Guide No. NS-G-2.13; Vienna.

IAEA – International Atomic Energy Agency (2010): Seismic Hazards in Site Evaluation for Nuclear Installations. Specific Safety Guide No. SSG-9, Vienna.

IAEA – International Atomic Energy Agency (2011): Power uprate in nuclear power plants: guidelines and experience/prepared within the framework of the Technical Working Group on Life Management of Nuclear Power Plants; Vienna.

JAVOR, E. et al. (2009): Introduction of Gd-2n fuel in Unit 4 in 2009. 8th Int. Conference on WWER Fuel Performance, Modelling and Experimental Support. Bulgaria.

KATONA, T. (2001): Seismic Safety Re-Evaluation and Enhancement at the Paks NPP. In: Workshop on the Seismic Re-evaluation of all Nuclear Facilities (Ispra, Italy, 2001); NEA/CSNI/R(2001)13/VOL4.

KATONA, T. & BAREITH, A. (2011): Response to the questions regarding seismic safety; provided by the HAEA in the context of the bilateral process; 04.11.2011.

KATONA,T. (2012): Answer to the questions Topic 1: Initiating events. EU Stress Tests Peer Review Meeting Luxembourg, (Power Point presentation); February 2012.

LAJTHA, G.; TÉCHY, Z. (NUBIKI, Hungary); ELTER, J.; TÓTH, E. (Paks NPP) (2009): Verification of the SAMG for Paks NPP with MAAP Code Calculations; OECD/NEA Workshop on "Implementation of Severe Accident Management (SAM) Measures" Schloss Böttstein Switzerland; October 26–28, 2009.

LÖRINCZ, K.D.; HORVÁTH, F. & DETZKY, G. (2002): Neotectonics and its relation to the Mid-Hungarian Mobile Belt. EU Stephan Mueller Special Publication Series, (2002) pp. 247–266.

MAGYARI, A.; MARSI, I.; THAMÓ-BOZSÓ, E. (2011): Indications of Late Pleistocene neotectonic and paleoseismic activity in the middle part of the Danube Valley (Pannonian Basin). XVIII INQUA Congress; Bern, Switzerland, 2011. www.inqua2011.ch/?a=programme&subnavi=abstract&id=2886&sessionid=55

MAROSI, S. & MESKÓ, A. (1997): Seismic Safety of the Paks Nuclear Power Plant, Akadémiai Kiadó,. ISBN 963-05-7452-7.

PAKS (2012): Seismic safety of the Paks NPP http://paksnuclearpowerplant.com/download/6179/Seismic%20safety%20of%20the%20Paks%20NPP.pdf.

PES (2004): Preliminary Environmental Study for LTE of Paks NPP.

PINCZES, J. (Paks NPP) (2009): Development of aging management and in-service inspection (concerning steam generators); steam generator corrosion.

PTS (2008): Fekete, T. & Tóth, P.: Summary of PTS Calculations. Budapest, 2008. Provided for the 14th bilateral meeting in 2008 by the Hungarian side.

RÓNAKY, J. (2012): Hungarian country presentation, Topic 1: Initiating events. EU Stress Tests Peer Review Meeting Luxembourg, (Power Point presentation); February 2012.

RSK – German Reactor Safety Commission (2010): Anforderungen an Dübelverbindungen in Kernkraftwerken, Stellungnahme; Anlage 1 zum Ergebnisprotokoll der 426. Sitzung der Reaktor-Sicherheitskommission am 20.05.2010.

RSK – German Reactor Safety Commission (2012): Recommendations of the RSK on the robustness of the German nuclear power plants; RSK recommendation 450th meeting on 26./27.09.2012.

SAM (2009): Overview of Paks Severe Accident Management Status; Provided for the 15th bilateral meeting in 2009 by the Hungarian side.

SG 1.26 Hungarian Atomic Energy Authority (HAEA) (2007): Guideline 1.26; Regulatory supervision of ageing management; Version: 2.

SG 4.12 Hungarian Atomic Energy Authority (HAEA) (2007): Guideline 4.12; Ageing management during operation of nuclear power plants; Version: 2.

SZOEKE, L. (2009): Capacity Upgrade at Paks Nuclear Power Plant. provided by the HAEA in the context of the bilateral process.

TÓTH, T. (2003): Folyófízi szeizmikus mérések, ELTE Geofizikai Tanszék, Budapest 2003 (Riverline seismic measurements); Unpublished PhD thesis, Geophysical Department, ELTE University, Budapest;144 pp.

TÓTH, L.; GYÖRI, E. & KATONA,T.J. (2009): Current Hungarian practise of seismic hazard assessment. In: OECD NEA Workshop: Recent Findings and Developments of Probabilistic Seismic Hazard Assessment (PSHA) Methodologies and Applications, Workshop Proceedings. Lyon, Franciaorczák, 2008.04.07–2008.04.09, p. 313-344 Paper NEA/CNSI/R; 2009. www.oecd-nea.org/nsd/docs/2009/csni-r2009-1.pdf.

TRAMPUS, P. (2010): PLiM for Paks; Nuclear Engineering International; 01 June 2010. University of Debrecen, Hungary http://www.neimagazine.com/story.asp?sc=2056542

UMWELTBUNDESAMT (2006): Report to the Austrian Government on Paks NPP Lifetime Extension – Environmental Impact Assessment; Revised Version – Vienna, June 2006. Unpublished.

UMWELTBUNDESAMT (2010): ROADMAP regarding LTE of Paks NPP. Follow-up questions to the 16th Bilateral Meeting between Hungary and Austria, Schloss Hernstein, Austria, November 22/23, 2010. Unpublished.

WENRA – Western European Nuclear Regulators' Association (2000): Nuclear Safety in EU Candidate Countries; October 2000.

10 GLOSSARY

A_m .... Acceleration Capacity (median)

AHNS .... Ad-hoc Group on Nuclear Security

AMP.................... Ageing Management Program

ASME .... American Society of Mechanical Engineers

BCC.................... Back-up Commend Centre

BDBA.................. Beyond Design Basis Accident

BOP.................... Balance-of-Plant

CAV .... Cumulative Absolute Velocity

CCF .... Common Cause Failure

CDF .... Core Damage Frequency

CDFM .... Conservative Deterministic Failure Margin

CRP.................... Copper Rich Precipitates

DACAAM .... Database established for monitoring ageing management

DBA .... Design Basic Accident

DBE .... Design Base Earthquake

DBT .... Design Basis Threat

ECC.................... Emergency Core Cooling

ECR.................... Emergency Control Room

EDG.................... Emergency Diesel Generator

EIA...................... Environmental Impact Assessment

ENIQ................... European Network for Inspection Qualification

ENSREG .... European Nuclear Safety Regulation Group

EOP.................... Emergency Operating Procedures

EU .... European Union

g .... Acceleration of free fall

Gd....................... Gadolinium

H_2 .... Hydrogen

HA .... Hydro Accumulator

HAEA.................. Hungarian Atomic Energy Authority

HCLPF................ High Confidence of Low Probability of Failure

HP .... High Pressure

I&C .... Instrumentation and Control

IAEA .... International Atomic Energy Agency

IAS...................... Information and Analytical Survey

ISI .... In-Service Inspection

KTA .... Kerntechnischer Ausschuss (Nuclear Safety Standards Commission)

LBLOCA .... Large Break Loss of Coolant Accident

LOCA .... Loss of Coolant Accident

LPECCS .... Low Pressure Emergency Core Cooling System

LRF .... Large Release Frequency

LTE .... Lifetime Extension

NPP .... Nuclear Power Plant

OBE .... Operating Basis Earthquake

Mn .... Manganese

Ni .... Nickel

PAR .... Passive Autocatalytic Recombiners

PCC .... Protected Command Centre

PDCA: .... Plan-Do-Check-Act

PGA .... Peak Ground Acceleration

PRISE .... Primary-to-Secondary Leakage

PSA .... Probabilistic Safety Assessment

PSHA .... Probabilistic Seismic Hazard Assessment

PSR .... Periodic Safety Review

PTS .... Pressurised thermal shock

PU .... Power Uprate

PWR .... Pressurised Water Reactor

RLE .... Review Level Earthquake

RPV .... Reactor Pressure Vessel

RW .... Radioactive Waste

SA .... Severe Accident

SAM .... Severe Accident Management

SAMG .... Severe Accident Management Guideline

SAR .... Safety Analysis Report

SBLOCA .... Small Break LOCA

SBO .... Station Black Out

SFM .... Spent Fuel Management

SFP .... Spent Fuel Pool

SG .... Steam Generator

SHA .... Seismic Hazard Assessment

SL .... Seismic Level

SMA.................... Seismic Margin Assessment

SPSA.................. Seismic Probabilistic Safety Assessment

SSC .... Systems, Structures and Components

T .... Temperature

TBq..................... Tera-Becquerel

TWE .... Through Wall Extent

UBA .... Umweltbundesamt

UHS.................... Ultimate Heat Sink

UMD .... Unstable Matrix Defect

UT....................... Ultra Sonic

WENRA .... Western European Nuclear Regulators´ Association

WWER or VVER. Water-Water-Power-Reactor, Pressurised Reactor originally developed by the former Soviet Union

ANNEX 1: FIGURES AND TABLES

Figure 3-1: Bridge structure between the bubble towers for reinforcing the reactor building structures (PAKS 2012)

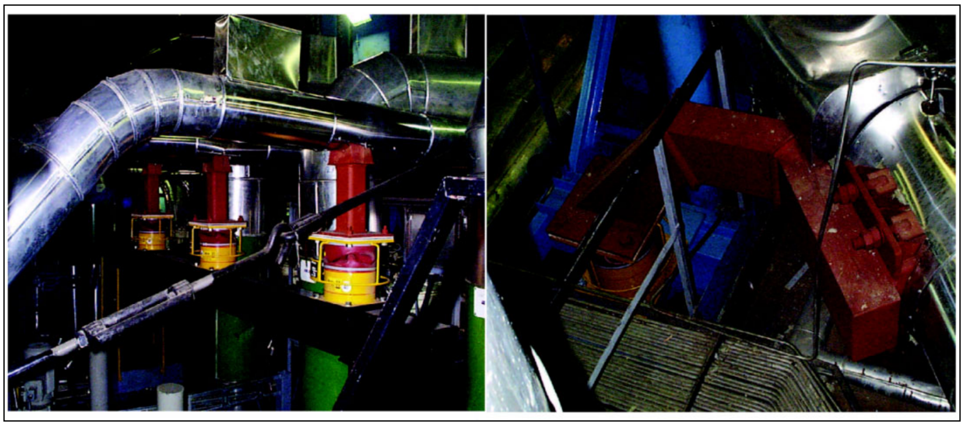

Figure 3-2: Application of viscous dampers (ELTER 2010)

Figure 3-3: Examples of easy-fixes (ELTER 2010)

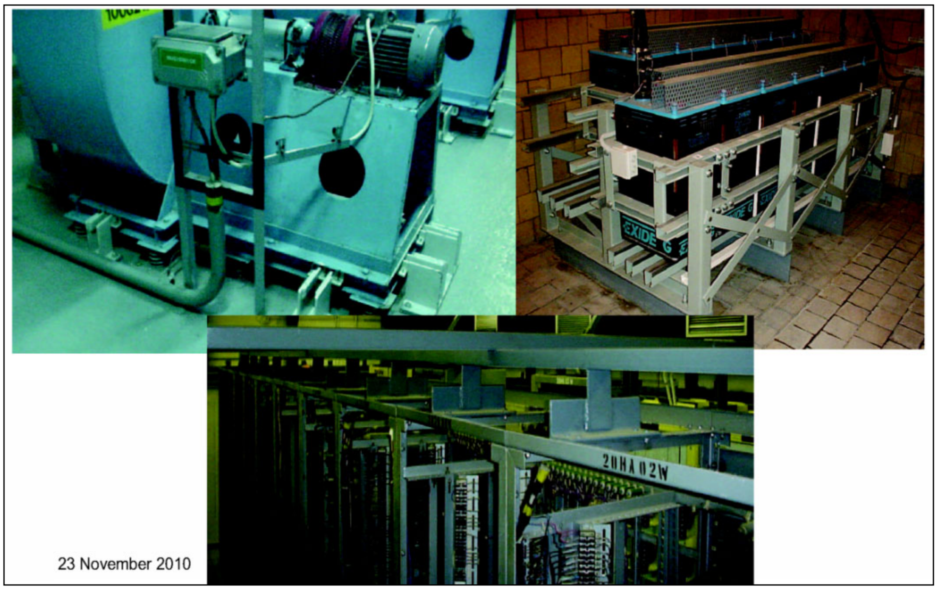

Figure 3-4: Examples of typical fixations (ELTER 2010)

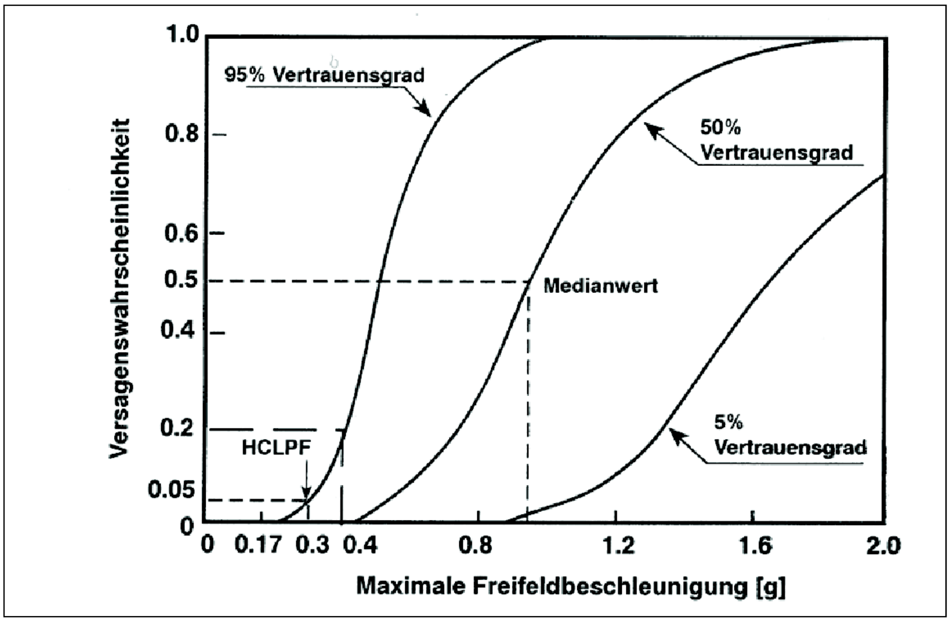

Figure 3-4: Schematic representation of failure curves (fragilities) for hypothetical component (BFS 2005)[30]

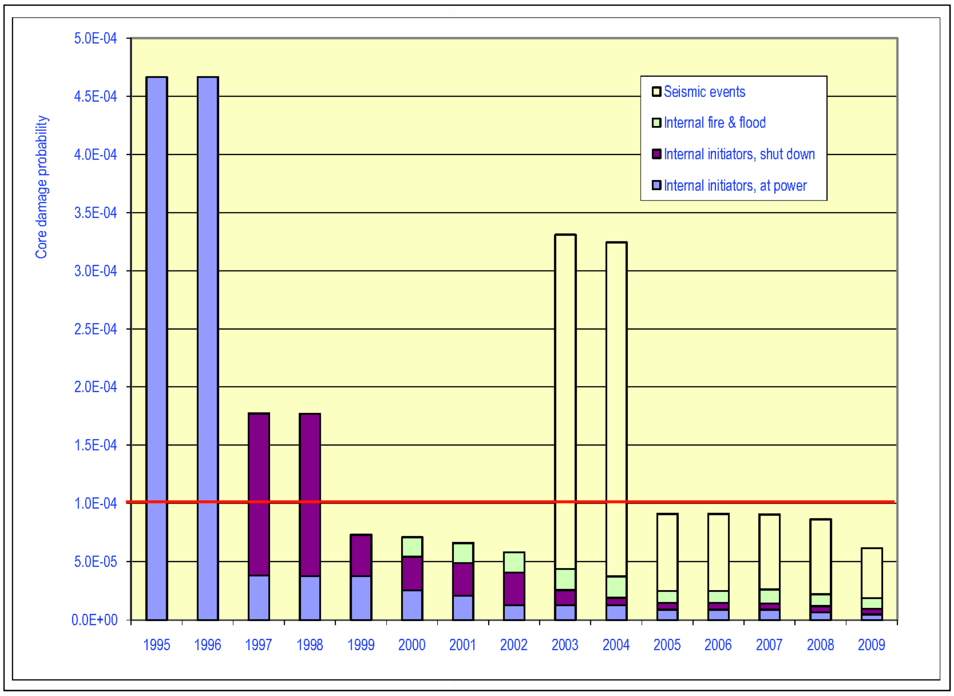

Figure 3-5: Core damage frequency for Paks NPP showing the relative contribution of different initiators (ELTER 2010)

[30] Translation of the legend: Vertrauensgrad = Confidence level; Medianwert = median value; maximale Freifeldbeschleunigung = free field maximum acceleration; Versagenswahrscheinlichkeit = failure probability.

Table 3-1: Summary of seismic upgrades at Paks NPP (ELTER 2010)

Qualification and upgrades	date	Volume of work
Electrical and I&C equipment	Easy fix, 1993-1995 -2002	450 t of steel structure added, batteries replaced, seismic instrumentation, Re-qualification of el. and I&C equipment
High energy pipelines of primary circuit and equipment	1997-1999	250 fixes (GERB viscous-dampers)
Building structure of the turbine and reactor hall	1999-2000	1360 t of steel structure added
Supporting frames of reactor building at the localization towers	2000-2001	300 t of steel structure added
Other classified pipelines of primary circuit and the equipment	1998-2000	760 fixes
Classified pipelines and equipment of secondary circuit, fixes of supporting steel structures in the turbine building	2000-2002	160 t of steel structure added
Classified pipelines of secondary circuit	2000-2002	1500 fixes
Other classified pipelines and equipment	2001-2002	80 fixes
Measures identified on the basis of seismic PSA	2002-	e.g. strengthening of all joints in the turbine building

Table 3-2: Accelerations ranges used for SPSA (BAREITH 2007)

Initiating Event	Acceleration Range (g), PGA	Mean Frequency (Event/Year)
SEIS1	0.07 - 0.10	$2.69 \cdot 10^{-3}$
SEIS2	0.10 - 0.15	$1.08 \cdot 10^{-3}$
SEIS3	0.15 - 0.22	$3.16 \cdot 10^{-4}$
SEIS4	0.22 - 0.32	$8.71 \cdot 10^{-5}$
SEIS5	0.32 - 0.48	$2.35 \cdot 10^{-5}$
SEIS6	0.48 - 0.70	$4.76 \cdot 10^{-6}$
SEIS7	0.70 - 1.00	$8.99 \cdot 10^{-7}$

Table 3-3: Distribution of core damage frequencies over acceleration ranges (HAEA 2011)

Acceleration range			Frequency of initiating event (1/year)	Core damage frequency (1/year)	Contribution (%)
Label	lower limit (g)	upper limit (g)			
SEIS1	0.07	0.10	$2.69 \cdot 10^{-3}$	$3.66 \cdot 10^{-8}$	0.08
SEIS2	0.10	0.15	$1.08 \cdot 10^{-3}$	$1.03 \cdot 10^{-6}$	2.39
SEIS3	0.15	0.22	$3.16 \cdot 10^{-4}$	$3.75 \cdot 10^{-6}$	8.69
SEIS4	0.22	0.32	$8.71 \cdot 10^{-5}$	$9.97 \cdot 10^{-6}$	23.14
SEIS5	0.32	0.48	$2.35 \cdot 10^{-5}$	$2.27 \cdot 10^{-5}$	52.57
SEIS6	0.48	0.70	$4.76 \cdot 10^{-6}$	$4.76 \cdot 10^{-6}$	11.03
SEIS7	0.70	1.00	$8.99 \cdot 10^{-7}$	$8.99 \cdot 10^{-7}$	2.09
Total:				$4.31 \cdot 10^{-5}$	100.00

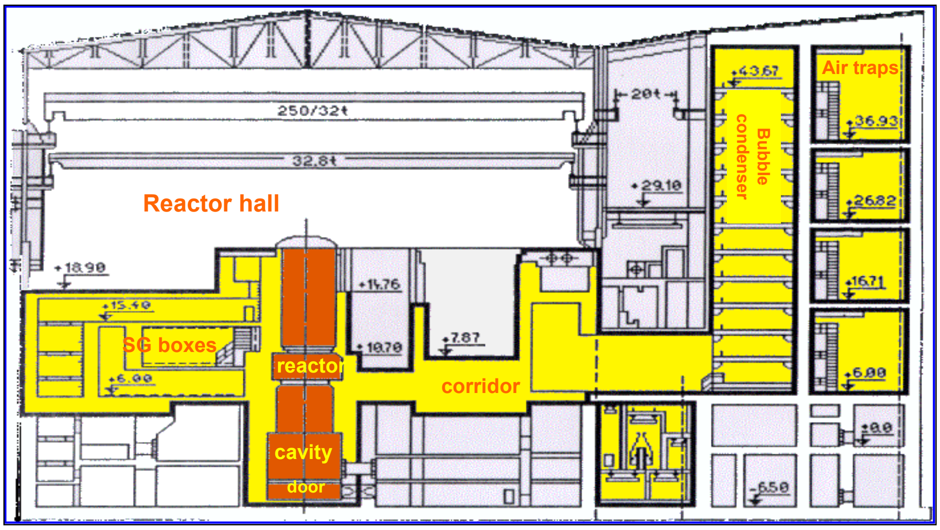

Figure 6-1: Containment (VVER 440, 213 type) with bubble condenser (Containment 2008).

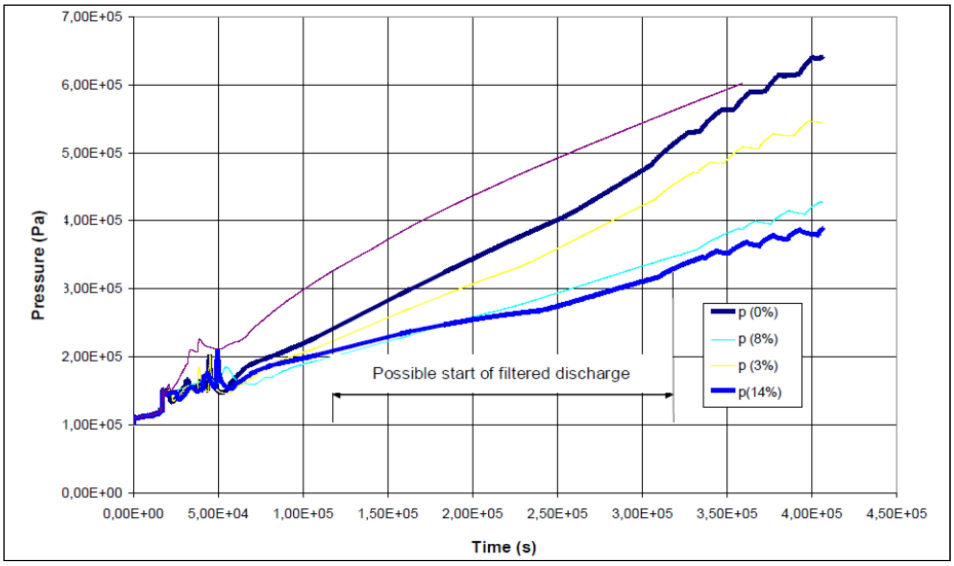

Figure 6-2: Pressure in the containment (reactor cavity is flooded) at different leakage rate values (HAEA 2011)

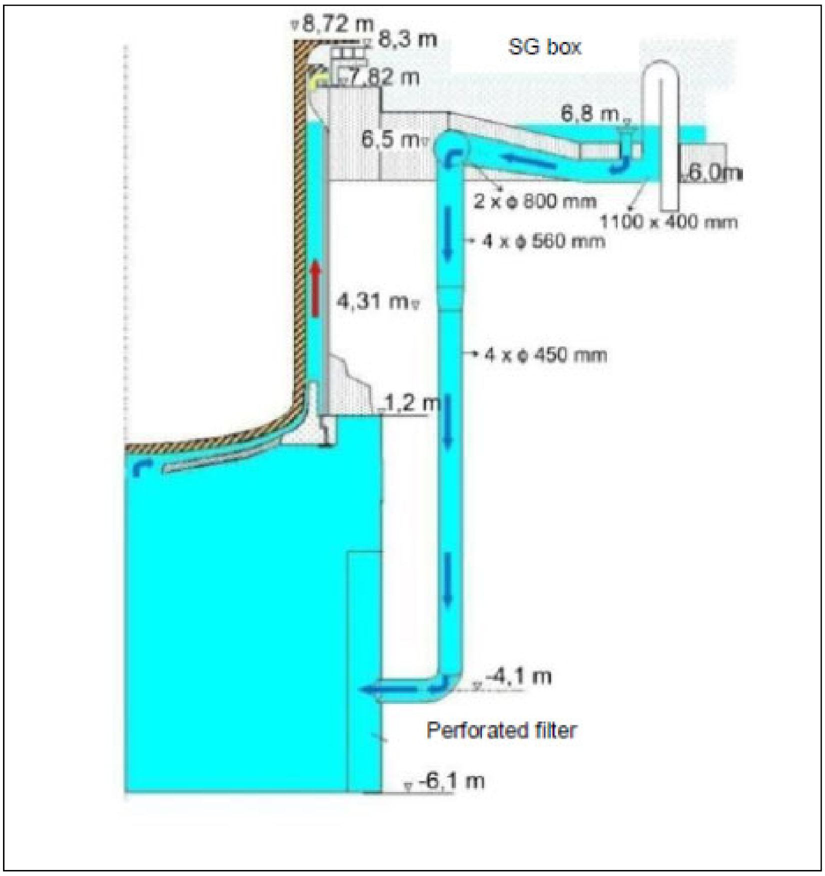

Figure 6-3: Scheme in principle of the external cooling of the reactor pressure vessel (HAEA 2011)

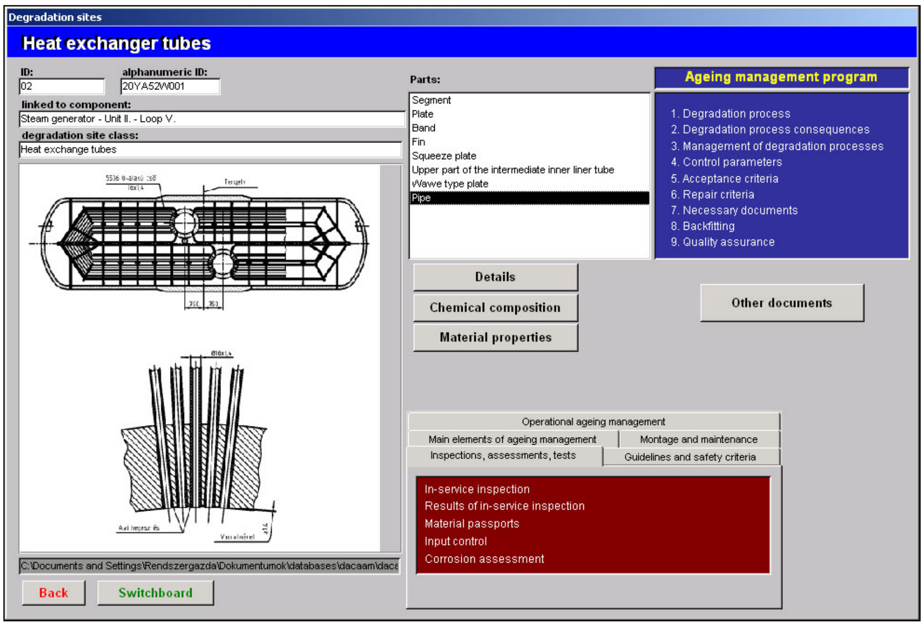

Figure 7-1: Critical component level AMP data for SG heat exchanger Tubes (PINCZES 2009)

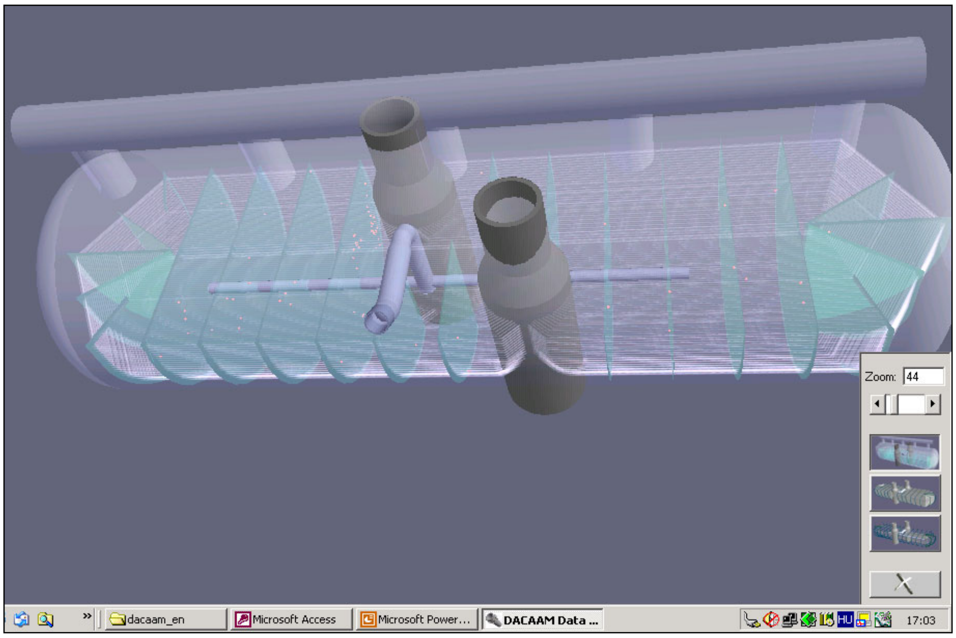

Figure 7-2: Maintenance history related data management and visualisation tools for SG heat exchanger tubes (PINCZES 2009)

Table 7-1: Correspondence between regulation concerning ageing management and in-service inspection (PINCZES 2009)

Aging Management	ISI
Hungarian Nuclear Safety Codes	
NSC (NSR) Volume 7. Definitions	
NSC (NSR) Volume 1. NPPR relevant Regulatory Procedures	
NSC (NSR) Volume 3. General requirements for the NPP design	
NSC (NSR) Volume 4. Safety requirements of the NPP operation	NSC (NSR) 4. Safety requirements of the NPP operation
Safety Guidelines	
Safety Guideline 4.12. Aging management during operation of nuclear power plants	Guideline 4.1. In-service inspection of the NPP equipments (Material Testing and Inspection)
Safety Guideline 1.26. Regulatory supervision of aging management	
Safety Guideline 3.13. Considerations of aging mechanisms during design	

ANNEX 2: COMPILATION OF THE DOCUMENTS PROVIDED BY THE HUNGARIAN SIDE

Documents provided by the Hungarian side in context of the 14th bilateral meeting

- Summary of PTS calculations; Tamás Fekete, Péter Tóth (AEKI); Budapest, October 2008; in this report referred to as (PTS 2008)
- Containment behaviour during DBA and BDBA events at Paks NPP. in this report referred to as (CONTAINMENT 2008)
- PTS Analysis of reactor pressure vessel, Guideline 3.17; Hungarian Atomic Energy Authority; Version 1; November 2005

Documents provided by the Hungarian side in context of the 15th bilateral meeting

- Aging Management: Development of aging and in-service inspection (concerning steam generators); steam generator corrosion
- Proposal of In-Vessel Corium Retention Concept for Paks NPP; Éva Tóth et al.; OECD/NEA-EC/SARNET2 workshop on In-Vessel Coolability 12-14 October 2009, Paris, France
- Uncertainty of the Level 2 PSA for NPP Paks; Gábor Lajtha, Attila Bareith, Előd Holló, Zoltán Karsa, Péter Siklóssy, Zsolt Téchy, VEIKI Institute for Electric Power Research, Budapest, Hungary
- Development of SAM strategy for Paks NPP on the basis of Level 2 PSA; Éva Tóth et al.; OECD/NEA workshop on Implementation of Severe Accident Management Measures (ISAMM-2009); 26-28 October 2009, Böttstein, Switzerland
- Influence of Power Uprate on Containment behavior and BDBA events at Paks NPP; Éva Tóth; Austrian-Hungarian Joint Commission; 30 November 2009, Budapest
- Overview of Paks Severe Accident Management Status; in this report referred to as (SAM 2009)
- Severe accident management measures planned and to be implemented (technical & organizational); Ferenc Medgyesy

Documents provided by the Hungarian side in context of the 16th bilateral meeting

- Insights of the seismic risk assessment and seismic upgrades, Paks nuclear power plant. József Elter; in this report referred to as (ELTER 2010)
- Capacity Upgrade at Paks Nuclear Power Plant. Larisza Szöke; in this report referred to as (SZOEKE 2009)
- Introduction of Gd-2n fuel in Unit 4 in 2009; 8th Int. Conference on WWER Fuel Performance, Modelling and Experimental Support. Bulgaria, 2009. Erica Jávor et al. in this report referred to as (JÁVOR 2009)
- Power Uprate Experience at the Paks NPP, Larisza Szöke, Lajos Hadnagy
- Report on the preparation works of the lifetime extension of the Paks Nuclear Power Plant

- First experiences with Gd-fuel assemblies in the Paks NPP; Tamás Parkó, Botond Beliczai; AER Symposium 2009.09.21-25.
- Analyses for licensing of new fuel types at NPP Paks; A. Keresztúri, S. Bogatyr, I. Nemes, A.V. Salatov; Finnish-Russian-Hungarian seminar on VVER-440 fuel technology and operational experience, Espoo, Finland; 19 – 20 October, 2004

Documents provided by the Hungarian side in context of the 17th bilateral meeting

- Pu and fuel: Status of the 2nd phase of fuel development; I. Nemes; Head, Section of Rph.; Paks NPP
- Response to the questions regarding seismic safety; Tamás János Katona, Attila Bareith; in this report referred to as (KATONA 2011).
- Response to the questions regarding pipeline vibration; Sándor Rátkai
- Verification of the SAMG for Paks NPP with MAAP Code Calculations; Gábor Lajtha, Zsolt Téchy (NUBIKI, Hungary); József Elter, Éva Tóth (Paks NPP); OECD/NEA Workshop on “Implementation of Severe Accident Management (SAM) Measures”; 26-28 October 2009 Böttstein, Switzerland; in this report referred to as (LAIJTHA 2009)

Documents provided by the Hungarian side in context of the 18th bilateral meeting

- CERES experiments calculation with the ASTEC code; Lajos Tarczal (Paks NPP), Gabor Lajtha (NUBIKI)
- Research Results in Support of In-vessel Corium Retention Program in the Paks Nuclear Power Plant; G. Ézsöl, G. Baranyai, L. Perneczky, L. Szabados MTA KFKI AEKI, Budapest (HU); 5th European Review Meeting on Severe Accident Research (ERMSAR-2012) Cologne (Germany), March 21-23, 2012
- Effects of 2nd phase of fuel development on neutron fluence in RPV wall; János Pinczés
- CERES test facility and test results; Éva Tóth
- SAM strategy and plant modifications at Paks NPP; Éva Tóth

ENVIRONMENT AGENCY AUSTRIA umweltbundesamt

Umweltbundesamt GmbH
Spittelauer Lände 5
1090 Wien/Österreich

Tel.: +43-(0)1-313 04
Fax: +43-(0)1-313 04/5400

office@umweltbundesamt.at
www.umweltbundesamt.at

ISBN 978-3-99004-206-9