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Stress test Follow-Up Actions

Issue Paper for Hungary

Authors:

Kurt Decker (University of Vienna)

Helmut Hirsch (cervus nuclear consulting)

Bojan Tomic (ENCO)

universität wien

cervus nuclear consulting

enco

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Table of Contents

1. Introduction

The EU post Fukushima Stress tests provided important insights into the robustness but also the vulnerabilities of individual NPP sites and units. Even during the performance of the Stress tests, having identified safety weaknesses, many plants embarked on modifications and safety improvements, in particular by adding mobile equipment. Following the completion of the Stress tests, all EU countries operating nuclear power plants prepared National Action Plans defining safety improvement measures and their implementation schedule. The National Action Plans addressed specific vulnerabilities found during the stress tests but also other elements, like safety improvements identified by other analyses or peer reviews.

Achieving and maintaining a high level of safety of NPPs in the neighbouring countries is of high interest to Austria. An important part of this is the understanding of and information concerning the implementation of the safety improvements, which are designed to rectify the vulnerabilities identified during the Stress tests, as well as other safety improvements. In order to identify the issues and safety improvements that are of highest relevance to Austria, the Federal Ministry for Agriculture, Forestry, Environment and Water Management engaged a group of Consultants (Project Team) to undertake an in-depth analysis of the Stress Tests reports (including operators' and regulators' reports), the Extraordinary CNS reports, the National Action Plans, but also some other sources like bilateral meetings and other previous discussions. The results of the analysis for Hungary are provided in the attached report.

Using the sources as described above, a set of safety issues and improvement measures of high interest for each of the neighbouring countries has been identified. Those issues and measures, following the same structure as used during the Stress Tests, are grouped into three categories:

- Topic #1: Initiating Events (earthquake, flooding and extreme weather)
- Topic #2: Loss of Safety Systems
- Topic #3: Severe Accident Management

For Hungary, there are several additional issues not directly related to the stress tests ("Topic #X") but also considered as important by the Austrian side. These issues cover topics which have been discussed during the Roadmap procedure concerning the lifetime extension of NPP Paks, and where some points remained open.

In each category relevant safety issues are listed. For each issue, the safety relevance and background information are provided. The information is, in general, taken from available reports and sources, and extended by the analyses of the Project Team. The Project Team provided its own estimates of the safety importance, as well as the expected schedule for the implementation. The latter (generally) reflects the schedules as provided by each country in the National Action Plan, though in some cases modified on the basis of perceived safety importance. Finally, the analysis of each of the safety improvements contains an entry called "To be discussed". In this entry, the specific details are summarized which are relevant for each specific safety issue and are considered to be of particular interest by the Project Team, and that are proposed to be discussed during bilateral meetings.

With the selection of safety issues and improvement measures, it is intended to open the discussion during the regular annual bilateral meetings with each of the neighbouring countries. It is expected that each of the safety issues and improvement measures will be followed up upon to their final implementation or resolution.

In order to assure that the safety improvements are discussed commensurate to their actual safety relevance, a categorisation of the issues has been proposed. With the analysis as described above, all the issues are grouped in 3 categories. The categorisation reflects the perceived safety importance of each issue or measure, but also the amount (and clarity) of information currently available. The three categories, in the increasing level of complexity are:

- Check list
- Dedicated presentation
- Dedicated workshop

The **"check list"** is assigned to the safety issues/improvement measures that are in general understood and specifics of which are either known or obvious. Considering this, it is expected that a short presentation is made describing the status and announcing the schedule for the completion of the issue/improvement measure.

The **"dedicated presentation"** is the next higher category. For issues/safety improvements in that category, it is expected that the countries will provide a dedicated presentation, where the relevant specifics of the issue or improvement measure will be highlighted in more details. This is expected to include e.g. the design concept, the specifics of the construction/implementation/analysis, or the planned operation of a modification.

For the issues of greatest safety significance but also for those of high complexity, or for the issues where the design solution is not known or many alternatives exist, the Project Team recommends that a **"dedicated workshop"** is organized. In this, the country would present all related details on the issue to enable the Austrian side to ask clarifying questions, to assure full understanding of the concept, details of installation/operation or any other element that is relevant for the issue/improvement measure. To increase the efficiency, some of the workshops are meant to address several related subjects as one set.

For presentations and workshops, the list in the "to be discussed" entry indicates the main (though not necessarily all) the elements that are of interest.

It is generally expected that each safety issue or improvement measure will remain on the agenda of bilateral meetings until the final completion and clarification. This does not mean that for each of the issues/improvements, a specific action (e.g. a workshop) would have to be made in each of the bilateral meetings. Rather, it is expected that in the course of the next several meetings all the issues will be addressed in accordance with a mutually agreed work plan.

2. Glossary

AC	Alternate Current
AFW	Auxiliary Feedwater
AHRS	Additional Heat Removal System
AM	Accident Mitigation
AMP	Ageing Management Program
ANSYS	Analysis System (finite element software)
ASME	American Society of Mechanical Engineers
ASTEC	Accident Source Term Evaluation Code
BD	Czech for Control Room (Bloková Dozorna)
BDB	Beyond Design Basis
BDBA	Beyond Design Basis Accident
BHB	German acronym for Operating Manual
BSVP	Czech for Spent Fuel Storage Pool (Bazén Skladováni Vyhořelého Paliva)
BMU	German Federal Ministry for the Environment
BWR	Boiling Water Reactor
CCW	Component Cooling Water
CW	Cooling Water
CDF	Core Damage Frequency
CERES	Cooling Effectiveness on Reactor External Surface
CEZ (ČEZ)	České Energetické Závody, Czech Electrical Utility
CH	Switzerland
CISRK	Czech for Central Radiation Monitoring System (Centrální Informačni Systém Radiačni Kontroly)
CNS	Convention on Nuclear Safety
CNS EOM	CNS Extraordinary Meeting
CRP	Copper-rich Precipitates
CS	Containment Spray
ČSN	Česká Norma
CST	Condensate Storage Tank
CVCS	Chemical & Volume Control System
CZ	Czech Republic
ČEPS	Czech Transition Grid (Česká Elektrická Přenosová Oustava)
DACAAM	Data Collection and Analysis for Ageing Management
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DE	Germany
DEC	Design Extension Conditions
DC	Direct Current
DG	Diesel Generator

E.ON	German Electrical Utility
EBO	Bohunice Nuclear Power Plant, Slovakia
EC	European Commission
ECC	emergency control centre
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
EDA	Power Plant Dalešice, Czech Republic
EDG	Emergency Diesel Generator
EDU	Dukovany Nuclear Power Plant, Czech Republic
EFW	Emergency Feedwater
EFWS	Emergency Feed Water System
EMO	Mochovce Nuclear Power Plant, Slovakia
EMS	European Macroseismic Scale
EnBW	Energie Baden-Württemberg AG, German Electrical Utility
ENSI	Swiss Federal Nuclear Safety Inspectorate (Eidgenössisches Nuklearsicherheitsinspektorat)
ENSREG	European Nuclear Safety Regulators Group
EOP	Emergency Operating Instructions
EPG	Emergency Power Generators
ERMSAR	European Review Meeting on Severe Accident Research
ES	Engineered Safeguards
ESCW	Essential Services Chilled Water
ESR	Electron Spin Resonance Dating
ESW	Essential Service Water
ETE	Temelín Nuclear Power Plant, Czech Republic
FWT	Feedwater Tank
GKN I	Neckarwestheim I Nuclear Power Plant, Germany
GKN II	Neckarwestheim II Nuclear Power Plant, Germany
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit, Germany
GPP	Gas Power Plant
HA	Hydro Accumulator
HAEA	Hungarian Atomic Energy Authority
HCLPF	High Confidence of Low Probability of Failure
HP	High Pressure
HŘS	Czech for Emergency Control Centre (Havarijní Řídící Středisko)
HU	Hungary
HVAC	Heating, Ventilation and Air Conditioning
HZSp	Czech for Fire Brigade of the NPP (Hasičský Záchranný Sbor Podniku)
IAEA	International Atomic Energy Agency
ICTS	Information and Communication Technology Services
IRS	Incident Reporting System
ISI	In-service Inspection
IZS	Czech for Integrated Rescue System (Integrovaný Záchranný System)

I&C	Instrumentation & Control
KBR	Brokdorf Nuclear Power Plant, Germany
KKB	Beznau Nuclear Power Plant, Switzerland
KKC	Czech for Emergency Coordination Centre (Krizové Koordinační Centrum)
KKE	Emsland Nuclear Power Plant, Germany
KKG	Grafenrheinfeld Nuclear Power Plant, Germany
	Gösgen Nuclear Power Plant, Switzerland
KKI-1	Isar I Nuclear Power Plant, Germany
KKI-2	Isar II Nuclear Power Plant, Germany
KKK	Krümmel Nuclear Power Plant, Germany
KKL	Nuclear Power Plant Leibstadt, Switzerland
KKM	Mühleberg Nuclear Power Plant, Switzerland
KKP I	Philippsburg I Nuclear Power Plant, Germany
KKP II	Philippsburg II Nuclear Power Plant, Germany
KKU	Nuclear Power Plant Unterweser, Germany
KRB B	Gundremmingen Nuclear Power Plant Unit B, Germany
KRB C	Gundremmingen Nuclear Power Plant Unit C, Germany
kV	Kilovolt
kW	Kilowatt
KWB A	Biblis Nuclear Power Plant Unit A, Germany
KWB B	Biblis Nuclear Power Plant Unit B, Germany
KWG	Grohnde Nuclear Power Plant, Germany
LFRS	Lead-Cooled Fast Reactors
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-site Power
LP ECCS	Low Pressure Safety Injection (within Emergency Core Cooling System)
LRF	Large Release Frequency
M	Magnitude
MCCI	Molten Corium Concrete Interaction
MCR	Main Control Room
MPa	Megapascal
MPLS WAN	Multiprotocol Label Switching Wide Area Network
MSK	Modified Mercalli Scale
NAcP	National Action Plan
ND	Czech for Emergency Control Room (Nouzová Dozorna)
NPP	Nuclear Power Plant
NRC	(US) Nuclear Regulatory Commission
OECD	Organisation for Economic Co-operation and Development
OECD/NEA	Nuclear Energy Agency of OECD
OSL	Optically Stimulated Luminescence Age dating
PAMS	Post-Accident Monitoring System
PAR	Passive Autocatalytic Recombiners

PC	Primary Circuit
PGA	Peak Ground Acceleration
PGAH	Peak Horizontal Ground Acceleration
PGAV	Peak Vertical Ground Acceleration
PSA	Probabilistic Safety Analysis
PSHA	Probabilistic Seismic Hazard Assessment
PSR	Periodic Safety Review
PTS	Pressurized Thermal Shock
PU	Power Uprate
PWR	Pressurized Water Reactor
RA	Radioactive
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RELAP	Reactor Excursion and Leak Analysis Program (simulation tool)
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RSK	Reactor Safety Commission (Advisory Body to German Federal Ministry for the Environment)
RWE	German Electrical Utility
RWST	Reactor Water Storage Tank
SA	Severe Accident
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SBLOCA	Small Break LOCA
SBO	Station Blackout
SCW	Service Circulating Water
SDSA	Steam Dump Station to Atmosphere
SFP	Spent Fuel Pool/pit
SFSP	Spent Fuel Storage Pool
SG	Steam Generator
SHA	Seismic Hazard Assessment
SiAnf	German Safety Requirements for Nuclear Power Plants
SK	Slovakia
SLO	Slovenia
SPSS	Secure power supply systems
SSCs	Structures, Systems and Components
StMUG	(Bavarian) State Ministry for the Environment
SÚJB	State Office for Nuclear Safety, Czech Republic
SUP	Safety Upgrade Program
SUSAN	Special Emergency System (Spezielles unabhängiges System zur Abfuhr der Nachzerfallwärme)
SW	Service Water
SWR69	German type of BWR

SWR72	German type of BWR
SZN	Czech for Safety Ensuring System (Systém Zajištění Bezpečnosti)
T_k	Ductile to Brittle Transition Temperature
TSC	Technical Support Centre
TVD	Czech for Essential Service Water (Technická Voda Důležitá)
UHS	Ultimate Heat Sink
UPS	Czech for Uninterruptible Power Supply (Nepřerušitelný Zdroj Elektrického Napájení)
V	Volt
VE	Czech for Hydroelectric Power Station (Vodní Elektrárna)
VVER	Water-Water-Energy-Reactor (reactor type of Soviet provenience)
WANO	World Association of Nuclear Operators
ZUNA	German acronym for AHRS

3. Summary of the Findings

SUMMARY TABLE
Stresstest Follow-Up Action: Issues for Monitoring, Hungary

Issue	Title	Safety importance	Follow-up	
			Action	Schedule
TOPIC 1: Initiating Events				
HU 1.1	Mitigation of liquefaction risks	High	Dedicated presentation	4Q/2014
HU 1.2	Assessment of Quaternary faults in the near-region of the site	High	Dedicated workshop	4Q/2015
HU 1.3	Measures to increase robustness of the plant	High	Dedicated presentation	4Q/2016
HU 1.4	Open questions relating to possible improvements of seismic design	Medium	Dedicated presentation	4Q/2015
TOPIC 2: Loss of Safety Systems				
HU 2.1	Installation of additional Diesel Generators (~2-3 MWe) to supply alternate electric	High	Dedicated workshop together with Dedicated presentation HU 2.2	4Q/2018
HU 2.2	Establishment of cross-connection between the essential power supply systems of the various units	High	Dedicated presentation together with Dedicated workshop HU 2.1	4Q/2018
HU 2.3	EDG cooling in case of loss of essential water system	High	Dedicated presentation	4Q/2015
HU 2.4	Ensuring black start possibility for gas turbine unit at Liter	Low	Check list	4Q/2017
HU 2.5	Alternate power supply for the traveling screens of the ESW pumps	High	Check list	4Q/2016
HU 2.6	Reliable power supply of the water wells	High	Dedicated presentation	4Q/2016
HU 2.7	External source for the make-up of spent fuel pool (SFP)	High	Dedicated presentation	4Q/2014
HU 2.8	Additional water source for the diesel-driven fire water pumps, and seismic qualification of the traveling screens	High	Dedicated presentation	4Q/2014
HU 2.9	Extension of alternative cooling	High	Dedicated presentation	4Q/2015
TOPIC 3: Severe Accident Management				
CZ/HU/SK 3.1	Stabilization of molten core of reactors of type VVER 440/213 (Bohunice, Dukovany, Mochovce, Paks)	High	Dedicated workshop[1]	1Q/2016
HU 3.2	Avoid long-term over-pressurization of containment	High	Dedicated presentation	4Q/2015
HU 3.3	Study of hydrogen generation and distribution in the reactor hall	High	Dedicated presentation	4Q/2014
HU 3.4	Measures against containment bypass via steam generator	High	Dedicated presentation	4Q/2014
HU 3.5	SAMGs to manage multi-unit accidents and simultaneous accidents in reactor and SFP	High	Dedicated presentation	4Q/2016
HU 3.6	Severe accident scenarios / PSA	Medium	Dedicated presentation	4Q/2017
TOPIC X: Outside Topics 1 - 3				
HU X.1	Reactor pressure vessel integrity	Medium	Dedicated presentation	4Q/2016
HU X.2	Ageing management	Medium	Dedicated presentation	4Q/2017

[1] For this Issue, a quadri-lateral workshop (between Czech Republic, Hungary, Slovakia and Austria) would be preferable. In case the Issue will be discussed in a bilateral framework, the questions will be revised to refer more specifically to what is relevant for each particular country.

3.1 Topic 1: Initiating Events (Earthquake, flooding and extreme weather)

HUNGARY	
Topic 1: Initiating events	
Issue No	**HU 1.1**
Title	Mitigation of liquefaction risks
Content	During the EU Stress Tests the Hungarian side provided information on a dedicated and technically outstanding study on liquefaction hazards. In that study quantitative assessments of 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 beyond the DBE. Soil liquefaction and related building settlement is expected to have major effects on inter-building connections and on the connections between cooling water wells and the plant. The issue is identified in the Hungarian National Action Plan (Working Group of the Hungarian Atomic Energy Authority, 2012,Task 4 and 5) on the implementation actions decided upon at the European Stress Tests. 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 are of particular interest to the Project team.
Safety relevance	Liquefaction is the dominating damage mode for ground motions beyond the DBE. Only a narrow safety margin is available.
Background	In 2011, a dedicated study of liquefaction hazard was performed by Hungarian experts in co-operation with experts from the Technical University of Berlin (e.g., Gyori et al., 2002). The results of that quantitative assessment were presented during the EU Stress Tests (Katona, 2012; Rónaky, 2012). This information may be used in the bilateral discussion by courtesy of the Head of the Hungarian Delegation to the European Stress Tests, Dr. József Rónaky. Permission was granted in a conversation following Hungary's country presentation. According to Katona (2012), the assessment of the safety margins against soil liquefaction reveal only narrow margins for the sediment layers between 10 and 20m beneath the site leading to the conclusion that liquefaction is expected as a dominating damage mode for seismic accelerations exceeding PGAH=0.25g. Soil liquefaction and consequent building settlement is expected to have major effects on inter-building connections. References: Gyori E, Toth L, Monus P, Zsiros T, Katona T, Site Effect Estimations with Nonlinear Effective Stress Method at Paks Npp, Hungary. In: EGS XXVII General Assembly. Nice, France, 2002.04.21-2002.04.26. Paper 4033.

	Katona, T. (2012). Answer to the questions Topic 1: Initiating events. EU Stress Test Peer Review Meeting Luxembourg, (Power Point presentation); February 2012 Rónaky, J. (2012). Hungarian country presentation, Topic 1: Initiating events. EU Stress Test Peer Review Meeting Luxembourg, (Power Point presentation); February 2012 Working Group of the Hungarian Atomic Energy Authority (2012). National Action Plan of Hungary on the implementation actions Decided upon lessons learned from the Fukushima Daiichi accident. http://www.ensreg.eu/node/685
To be discussed	The requested presentation should include the following information: ➢ Type of the safety relevant structures, systems and components endangered by liquefaction. ➢ Measures envisaged to increase the robustness of these SSCs. ➢ Additional studies decided to reduce the uncertainties of the liquefaction hazard assessment.
Safety importance	High
Expected schedule	Short term
Follow-up	Dedicated presentation

HUNGARY	
Topic 1: Initiating events	
Issue No	**HU 1.2**
Title	Assessment of Quaternary faults in the near-region of the site
Content	Seismic hazard assessments for the NPP Paks included serious efforts to identify capable faults in the site vicinity. Fault mapping by high-resolution seismic reflection profiles revealed several faults, which offset Quaternary sediments. These faults need to be classified as “capable” according to international practice and IAEA documents (IAEA, 2003; IAEA, 2010). Stress Test documents and the material supplied during the Hungarian-Austrian bilateral meetings did not sufficiently clarify whether these faults have been studied further. The information that has been provided to the Austrian side so far does not specify whether or not an adequate and systematic effort was undertaken to constrain the youngest fault slip history and assess paleo-earthquakes, which may have occurred at these faults. It remains further unclear whether fault assessments (e.g., by paleoseismological methods or geodetic data) are included in the present seismic hazard assessment, or this analysis solely relies on historical and instrumental earthquake data. IAEA (2010) strongly recommends including fault-specific data in the assessment of intraplate areas in order to overcome the deficiencies of the earthquake database. The topic appears particularly important with respect to the location of the plant in the vicinity of the Mid Hungarian Fault Zone, which has been identified as an active seismogenic source by investigations of Hungarian geoscientists.
Safety relevance	The systematic assessment of Quaternary faults in the near region and site vicinity, and the parameterisation of slip history, youngest slip events and slip velocity are of utmost importance for the reliability of seismic hazard assessment. According to IAEA (2010) potential capable faults in the site vicinity of existing NPPs should be investigated in detail to provide a basis to “decide conclusively that the fault is not capable”. The question on Quaternary fault assessment has previously been addressed at both, the bilateral meetings at Herrnstein (2010) and Pecs (2011). A clarification of the issue should therefore be reached in near future.
Background	The seismic hazard issue was the topic of a dedicated presentation at the 16th Hungarian-Austrian bilateral meeting in 2010 (Elter, 2010). During this meeting a number of questions on technical details arose, which were forwarded to the Hungarian side (UBA, 2010). The Hungarian side provided written response the outlined questions timely before the 17th bilateral meeting in 2011 (Katona, 2011). During the preparation of the agenda for the 17th bilateral meeting both, the Hungarian and Austrian side decided not to discuss seismic issues at the upcoming meeting as the Hungarian experts were expected to provide additional information on the seismic safety of the Paks NPP for the Stress Tests. These data include 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. During this review supplementary data and information was provided to the Austrian experts. 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 bilateral discussion upon Austrian request. Permission was granted in a conversation following Hungary's country presentation. The whole process, however, did not resolve the issue. Open questions remain in the following three topics: **Sseismic hazard assessment** The 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, the design base earthquake level of PGA_H=0.25g and PGA_V=0.20g for the occurrence probability of 10^{-4}/year was established in 1996. Those values, corresponding to the SL-2 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, 2011; 2012). The results of the study were published by Tóth (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. **Assessment of Quaternary faults in the near-region and site vicinity** The most important questions in the previous bilateral discussion addressed the use of geological and microseismic data in the 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 convincing evidence for capable faults in the near-region and the site vicinity of the plant (e.g. Tóth, 2003: Line PA-3b/93as; Horvath, 2004; Magyari 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 site vicinity and near-regional faults. Response to this key information request of the Austrian side was received

	before the 17th bilateral meeting in 2011 (Katona, 2011). As the information supplied in this document was not regarded sufficiently clear it was decided to track the issue during the EU Stress tests. Information obtained through the EU Stress Tests is included in HAEA (2011), ENSREG (2012), Katona (2012), and 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 account for the microseismic data. It was further claimed that paleoseismological investigations have been performed. No details of such studies were presented. The only reference to paleoseismological investigations included in Katona (2012) refers to work by Magyari Árpád 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 near-regional and site vicinity faults. The presentation during the Peer Review meeting at Luxembourg Stress Tests further shows 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 (0.35g) 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. **New seismic hazard assessments in the course of the siting of Paks 5&6** Requested information further addresses 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 (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, 2003). 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 evidences 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 the plant of 20 years will be considered in seismic risk assessment. **Assessment of the information obtained so far** The information obtained on seismic hazard assessment and the assessment of Quaternary faults in the near-region of the site is not sufficiently clear. It is evident that seismic hazard assessments in the 1990ies 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 studied further 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 (<5km) of faults which offset Quaternary sediments and therefore are to be classified as "capable" in the sense of IAEA (2010). Some of the mapped faults parallel the Mid Hungarian Fault Zone, which has been identified as an active seismogenic source by investigations of Hungarian geoscientists (Lörincz, 2002; Horvath, 2004; Bus, 2009). At this background a systematic assessment of Quaternary faults and the parameterization of slip history, youngest slip events, fault geometry and slip velocity is of utmost importance for the reliability of seismic hazard assessments. References: 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: 343-352. Elter, József (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/node/398 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/node/362 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 (2003): Site evaluation for Nuclear Installations. Safety Requirements, No. NS-R-3, Vienna 2003. http://www-pub.iaea.org/MTCD/Publications/PDF/Pub1177_web.pdf IAEA (2010): Seismic Hazards in Site Evaluation for Nuclear Installations. Specific Safety Guide No. SSG-9, Vienna 2010. http://www-pub.iaea.org/MTCD/publications/PDF/Pub1448_web.pdf 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 Test Peer Review Meeting Luxembourg, (Power Point presentation); February 2012 Lörincz, K.D., Horváth, F. & Detzky, G. (2002): Neotectonics and its relation to the Mid-Hungarian Mobile Belt. EU Stephan Mueller Special Publication Series, 247-266. Marosi, S. & Meskó, A. (1997). Seismic Safety of the Paks Nuclear Power Plant, Akadémiai Kiadó, 1997. ISBN 963-05-7452-7 Magyari 2011). Indications of Late Pleistocene neotectonic and paleoseismic activity in the middle part of the Danube Valley (Pannonian Basin). XVIII INQUA Congress, 21-27 July 2011 Bern. http://www.inqua2011.ch/?a=programme&subnavi=abstract&id=2886&sessionid=55 Rónaky, J. (2012). Hungarian country presentation, Topic 1: Initiating events. EU Stress Test Peer Review Meeting Luxembourg, (Power Point presentation); February 2012 Tóth, T. (2003). Folyófízi szeizmikus mérések, ELTE Geofizikai Tanszék, Budapest 2003, 144 pp. (Riverline seismic measurements); Unpublished PhD thesis, Geophysical Department, ELTE University, Budapest 2003, 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 UBA (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.

To be discussed	The Project team therefore renews and extends the information request regarding seismic hazard assessment and the analysis of capable faults. The requested workshop with Hungarian geoscientists should address the following questions: ➢ Reflection seismic data acquired during the seismic hazard assessment programs in the late 1990ies and early 2000nds identified several Quaternary faults in the near-region of Paks. Have these faults been further investigated with proper methodologies, and have data that adequately describe the faults been integrated into the seismic hazard model? ➢ Did the finding of Quaternary faults and potentially capable faults in the site vicinity of the NPP trigger additional research to assess faults capability with proper methodologies such as the ones suggested by IAEA (2010) in order to constrain their slip histories, the youngest slip events, fault geometries and slip velocities? ➢ Have the results of the continued microseismic monitoring been used to identify active faults in the NPP's near-region and site vicinity? ➢ 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 analyze 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). 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 latest PSHA account for Ove Arup's modelling results? ➢ The latest PSHA is based on 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? ➢ What kind of measures (geological/geophysical data acquisition, hazard reviews, novel hazard assessments etc.) will be required for the siting and licensing of Paks 5&6? If already decided: what is the current status of such investigations?
Safety importance	High
Expected schedule	Medium term
Follow-up	Dedicated workshop

HUNGARY	
Topic 1: Initiating events	
Issue No	**HU 1.3**
Title	Measures to increase robustness of the plant
Content	The Hungarian NAcP lists several possible measures to increase the seismic robustness of Paks NPP. Envisaged measures are concerned with the fire brigade building, protection of the demineralised water tanks, the 400kV and 120kV switchyard, seismic qualification of parts of the essential service water system, automatic reactor shut-down (in the framework of the reconstruction project of earthquake instrumentation) and qualification of the on-site shelters.
Safety relevance	Appropriate measures of seismic design increase the robustness against seismic loads beyond the design basis. With higher safety margins, the chances for the plant to withstand an earthquake of unexpected strength are increased.
Background	The results of the assessments concerning the necessity of measures envisaged to increase robustness of the plants against earthquakes – as presented in chapter 2.2.4. of HAEA (2011) – and their respective implementation, are of interest to the Project team. Not all the measures listed in chapter 2.2.4 are addressed here; a selection of those seen as most relevant has been performed. These measures have also been identified in the Hungarian National Action Plan (Working Group of the Hungarian Atomic Energy Authority, 2012, Task 2, 3, 9, 11, 14, and 47): Action 2: *Interventions to protect the personnel and equipment in the fire brigade barrack […]* Action 3: *The demineralised water tanks of Installation II […] are located in the direct vicinity of the medical and laboratory building. The walls of the building shall be seismically qualified and, if necessary, reinforced […]* Action 7: *Automatic shutdown of the main condenser coolant pumps shall be provided when the condenser pipeline is damaged due to earthquake or other reason. It shall be ensures that the pipeline trenches are applicable to receive and drain the discharged water […]* Action 9: *In the frame of the construction project of seismic instrumentation […], the question of automatic shutdown shall be revisited.* Action 11: *Protection of the not seismically reinforced 400 kV and 120 kV substations shall be evaluated and increased if necessary.* Action 14: *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 jeopardize the ultimate heat sink function […]* Action 47: *Seismic qualification of the on-site shelters not yet qualified shall be performed and non-earthquake resistant equipment in the shelters shall be improved […]* Deadline for Action 9 is end of 2012, for Action 11 end of 2014, for Actions 2, 3, 7 and 14 end of 2015 and for Action 47 – end of 2016.

	References: 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/node/362 HAEA (2012). National Action Plan of Hungary on the implementation actions Decided upon lessons learned from the Fukushima Daiichi accident. http://www.ensreg.eu/node/685
To be discussed	The Project Team requests the following information: ➢ What were the results of the investigations performed in the context of the NAcP Actions listed above? ➢ Which decisions have finally been taken with respect to these investigations? ➢ Which back fitting measures are envisaged accordingly, and what is the time schedule for their implementation?
Safety importance	High
Expected schedule	Medium term
Follow-up	Dedicated presentation

HUNGARY	
Topic 1: Initiating events	
Issue No	**HU 1.4**
Title	Open questions relating to possible improvements of seismic design
Content	A number of questions regarding seismic design have been raised in the course of the **Paks Roadmap** process (UBA, 2012) and were noted for further treatment in the framework of the present project. These questions concern mounting of anchor bolts, superposition of operating conditions at low-power and shutdown with a design basis earthquake, and operator actions in case of an earthquake.
Safety relevance	The points mentioned here are potentially relevant for seismic safety and might require additional measures to increase robustness.
Background	**Inspections of already installed anchor bolts (UBA, 2012)** 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 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. **Occurrence of a design basis earthquake during low-power and shutdown operation (UBA, 2012)** As pointed out in HAEA (2011) the occurrence of an earthquake during plant states of short duration (e.g. displacement of the refuelling and hoisting machines during refuelling) was not assumed in the context of the determi-nistic 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 concer-ning 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 earth-quake 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). **Safe shutdown after a design basis earthquake (UBA, 2012)** 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.

	Katona & Bareith (2011) further state 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 in the course of deterministic safety analyses that operator actions are successfully performed within short time periods (e.g. within 30 minutes). References: 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. 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/node/362 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. 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. UBA (2012). Roadmap of Paks NPP Lifetime Extension (LTE) – Summary of Current Status and Open Questions of the Bilateral Meetings between Austria and Hungary. Rep-0402. http://www.umweltbundesamt.at/aktuell/publikationen/publikationssuche/publikationsdetail/?pub_id=2015
To be discussed	The following points should be clarified in order to assess whether additional measures are required to improve seismic design: ➢ The necessity to perform inspections of already installed anchor bolts to check whether they have been mounted correctly. ➢ What are the possible consequences of a superposition of operating conditions during low-power and shutdown operation of short duration with a design basis earthquake? ➢ Does the deterministic safety case rely on successful operator actions within short time periods after an earthquake (e.g. within 30 minutes)?
Safety importance	Medium
Expected schedule	Medium term
Follow-up	Dedicated presentation

3.2 Topic 2: Loss of Safety Systems

HUNGARY	
Topic 2: Loss of safety systems	
Issue No	**HU 2.1**
Title	Installation of additional Diesel Generators to supply alternate electric power (for each unit)
Content	In addition to the main safety related DGs (3x100% per unit), Paks NPP is equipped with 4 so-called severe accident diesel generators (300 kW), to supply electrical power to measurement and control systems, but none to active safety systems (pumps). Therefore, additional DGs to supply power to safety consumers is justified. The capacity of the DG shall be such to enable supply of equipment (2 to 3 MW is envisaged/expected)that would be essential for prevention and/or control of severe accidents sequences.
Safety relevance	In the case of total blackout, the severe accident mobile diesel generators (300 kW) available at the each Unit are capable to supply only the I&C and some essential communication/lighting and operation of selected controls (valves). These diesel generators are not capable to supply electrical power to safety systems and in particular to the essential service water pumps, which are essential for the operation of the heat sink. Therefore, the installation of additional power supply will significantly increase prevention and accident management abilities.
Background	Additionally to the safety related electrical power supply system, one 300 kW severe accident diesel generator is available for each unit. These DGs are installed on trailers and can be hauled; when not in use they are stored in an earthquake proof building on the site of the nuclear power plant. The severe accident diesel generators, according to their design basis, are capable to supply electrical power to the instrumentation, monitoring and intervention systems, which are needed for mitigating the consequences of the severe accident (e.g. pressure reduction of the primary circuit, flooding of the reactor cavity). These DGs are not capable to supply electrical power to safety supply systems and to the essential service water pumps, therefore in order to manage accident situations the establishment of an additional, diverse diesel generator is planned. The capacity of the diverse accident diesel generator has to be determined in such a way that it has to be capable to supply electrical power to the safety required consumers, as appropriate for specific sequences. The number and capacity of the diverse accident diesel generators still needs to be determined. The NAcP report of Hungary indicates the deadline for the installation of new DGs as 15. December 2018. The presentation on the status of NAcP activities in Hungary at the regular bilateral meeting on November 18/19, 2013, in Budapest indicated that 2 DGs for the four units of Paks NPP, of 6 MW capacity each, will be installed, one DG having enough capacity to supply power to all four units (including the SFPs). The current deadline for the implementation remains 2018, but this will be very difficult to achieve due to the very complicated procurement process.

	References: HAEA (2011). National Report of Hungary on the Targeted Re-assessment of Paks Nuclear Power Plant, HAEA, December 2011. http://www.ensreg.eu/node/362 ENSREG (2012). Peer Review report on EU Stress Tests for Hungary, ENSREG, April 2012. http://www.ensreg.eu/node/398
To be discussed	The presentation should describe in more detail the safety concept and design of the proposed measure and the following issues should be discussed: ➢ What is the implementation schedule and progress to date, including regulatory approval, assuming that the date in the NAcP is still valid? ➢ Which configurations of possible power supply connections are envisaged? ➢ What will be the technical characteristics of the diverse AM DGs (including DG cooling, fuel, autonomy, etc)? ➢ How will they be protected against external hazards? ➢ Which consumers would they supply a) prior to core damage and b) in case of SA? ➢ What will be the safety margin gained in different design and beyond design basis accident sequences?
Safety importance	High
Expected schedule	Long term
Follow-up	Dedicated workshop (together with Dedicated presentation HU 2.2)

HUNGARY	
Topic 2: Loss of safety systems	
Issue No	**HU 2.2**
Title	Establishment of cross-connection between the essential power supply systems of the various units
Content	If the total loss of electrical power does not occur at all units simultaneously, and the generators operating on in-house load level of certain units or their safety power supply remain operable, then it is possible to supply electrical power between the units, both between normal and reserve 6 kV service systems. Cross-connections will be installed allowing the 6 kV bars to be supplied from any operating main generator (in house load mode) or operable emergency diesel generator of any unit to any other unit.
Safety relevance	Availability of power for safety systems has a major impact on prevention as well as mitigation of accidents. For sites with multi units (in case that not all units on a site are affected) numerous power sources exist. If there is a possibility to cross connect those sources the level of defence will be strengthened. Therefore, the realization of this modification and the development of the related operating instructions is a substantial increase in the power supply reliability.
Background	If the total loss of electrical power does not occur on each unit simultaneously, and the generators operating on in-house load level of certain units or their safety power supply remain operable, then the easiest solution is to provide electrical power through the 400 kV substation, if it is operable. The National Report of Hungary on the Targeted Re-assessment of Paks Nuclear Power Plant presents the following: A solution in principle exists to supply electrical power between all four units , both between normal and reserve 6 kV systems. The NAcP indicates that the procedure to enable such a connection shall be in place by 31. July 2013 The connection between the units from the emergency diesel generators has not yet been realized. Since alternative measures play a significant role in the prevention of the evolution of severe accident scenarios, therefore the list of corrective measures include the realization of this option and the development of the relating operating instructions. Consequently, the 6 kV safety systems of any unit would be supplied from any operating emergency diesel generator. The NAcP indicates that the arrangement necessary for connecting DGs needs to be in place by 15. December 2015. The ENSREG Peer Review Report states that a feasibility study has to be prepared to improve the potential supply routes between the 6 kV safety systems of the units, in order to find the potential solution ensuring power supply to the 6 kV safety system of each unit from each EDG without the use of the external grid. The modifications required by the study have to be realized. References: HAEA (2011). National Report of Hungary on the Targeted Re-assessment of Paks Nuclear Power Plant, HAEA, December 2011. http://www.ensreg.eu/node/362 ENSREG (2012). Peer Review report on EU Stress Tests for Hungary, ENSREG, April 2012. http://www.ensreg.eu/node/398

To be discussed	The presentation should describe in more detail the safety concept and design of the proposed measure and answer the following questions: ➢ Is such a modification still maintaining the original design concept of the plant? ➢ What is the implementation schedule on both actions, and progress to date i.e. is the NAcP action 26 completed as it should have been considering the deadline in the report), including regulatory approval? ➢ What connection configurations are envisaged? Will the connection be automatic or manual? ➢ How is the separation of the safety trains being maintained? If all trains could be cross connected, then the short circuit might disable several trains? ➢ How is the overload protection being assured? ➢ What will be the safety margin gained in different design and beyond design basis accident sequences?
Safety importance	High
Expected schedule	Long term
Follow-up	Dedicated presentation (together with Dedicated workshop HU 2.1)

HUNGARY	
Topic 2: Loss of safety systems	
Issue No	**HU 2.3**
Title	EDG cooling in case of loss of essential water system
Content	Installation of the alternate EDG cooling route from the fire water system, for at least one EDG per unit.
Safety relevance	In the current design, essential service water is used for the cooling of the diesel generators. With a loss of ESW, EDGs become inoperable. In order to ensure cooling of EDGs, an alternate cooling system will be planned and installed, relying on the fire water system, pumps of which are driven by dedicated diesel engines. With this, independence of the EDGs from the ESW will be achieved, assuring availability of EDGs for a wider scope of sequences.
Background	The Hungarian National report highlights the special configuration at the Paks NPP whereby the electric power supply of safety systems (the operation of the emergency diesel generators) requires essential service water, while the essential service water system is supplied, in case of loss of offsite power at all four units, by the EDGs. However, the ENSREG Peer review report states that in addition to the cooling with essential service water, the EDGs could be cooled by the fire hydrant and the connecting point is easy accessible to get cooling quickly. It is understood that at one DG per unit a fire water connection has already been installed (as demonstrated during the visit of the Peer review team in April 2012). The NAcP indicates that the connection for one DG at each of the units will be in place by 15. December 2013. It is nevertheless of interest to understand whether such connections will be installed for other DGs References: HAEA (2011). National Report of Hungary on the Targeted Re-assessment of Paks Nuclear Power Plant, HAEA, December 2011. http://www.ensreg.eu/node/362 ENSREG (2012). Peer Review report on EU Stress Tests for Hungary, ENSREG, April 2012. http://www.ensreg.eu/node/398
To be discussed	The issues of interest include: ➢ The environmental conditions (extreme weather) in which this cooling arrangement will remain operable. ➢ The time duration of establishing the cooling connection, after the EW system is lost. ➢ The safety benefits of this modification, in terms of which sequences that would see safety benefits. ➢ Consideration of extending the cooling connection from fire water to all DGs at the site, pros and cons and analysis undertaken in this respect.
Safety importance	High
Expected schedule	Medium term
Follow-up	Dedicated presentation

HUNGARY	
Topic 2: Loss of safety systems	
Issue No	**HU 2.4**
Title	Ensuring black start possibility for gas turbine unit at Litér
Content	At present, the gas turbine at Liter needs a grid for its start. With the installation of a dedicated DG at Litér, the unit could start with offsite grid unavailable and supply electricity to Paks NPP.
Safety relevance	In a case of general unavailability of power distribution systems, dedicated supply could be achieved from fast starting gas units. If the unit would have a black start capability, it would be available to supply the power to Paks NPP in a case of unavailable grid. Accordingly, electrical supply routes were established from Litér through the 400 kV transmission line, including black start after the total loss of electric power supply. However, this unit is 90 km away and if the unavailability of the grid is caused by natural destruction, this would not help much.
Background	A power supply route can be realized from the Gas Turbine Plant located in Litér to the 6 kV electric main distributor of the nuclear power plant through the 400 kV transmission line between Paks and Litér. Based on accomplished tests the arrangement of the dedicated supply route and the required switching operations can be performed within one hour both at the system controller and at the plant. The black-start simulator training of the electric operators includes the realization of the dedicated switching arrangement required by the operating instruction. Currently, the Gas Turbine Plant in Litér does not have an autonomous power source (i.e. own diesel generator); in the frame of this re-assessment the licensee formulated a corrective action in this regard. The NAcP indicates that the deadline for this action is 15. December 2014. Reference: HAEA (2011). National Report of Hungary on the Targeted Re-assessment of Paks Nuclear Power Plant, HAEA, December 2011. http://www.ensreg.eu/node/362
To be discussed	The issues of interest include: ➢ Why was the Litér unit selected for the dedicated power supply to Paks? Aren’t there plants that are located closer to the Paks site, like e.g. Dunamenti plant? ➢ How is the coordination between Paks NPP and Litér operators achieved? In a case of large destruction (seismic event) how would the communication be maintained? Has Litér a special procedure instructing operators to restore Paks supply first? ➢ What is the time interval needed to establish the connection and supplying power to Paks NPP? Are there any manual actions needed in this respect?
Safety importance	Low (limited sequences where this would add value)
Expected schedule	Long term
Follow-up	Checklist

HUNGARY	
Topic 2: Loss of safety systems	
Issue No	**HU 2.5**
Title	Alternate power supply for the traveling screens of the ESW pumps, and seismic qualification of the traveling screens
Content	The essential service water system traveling screens will be provided with an alternate (safety related) power supply to enable operation also in a case of loss of main power supply.
Safety relevance	The loss of the electrical power will result in the termination of cleaning of the screens in the inlet line of the ESW. Each of the three screen systems requires regular cleaning. Despite the fact that lack of cleaning would not have immediate effects, no cleaning during a certain time period would ultimately result in ESW pump suction being lost. With this, all the consequences of the plant losing the ultimate heat sink will occur. With this modification, the power will be assured, so that the ESW will continue to operate. In a case of the seismic event, the traveling screen might be damaged/destroyed limiting the ability to provide water supply to the SWS.
Background	The heat sink function is realized by the chain of several systems, the ultimate element of which is the Danube. The heat removal from the fuel may be lost, if the connection between the cooling systems of the plant and the Danube would be lost. The principal element of this connection is the essential service water system (ESWS) that is installed according to the safety principle of threefold redundancy. Two cooling water pumps in each of the three redundant lines of the ESWS (i.e. altogether 6 pumps/installation) suck the cooling water from the cold water channel through a pre-screening filter. The loss of the electrical power may cause such a problem that the cleaning of the screens (i.e. bar screen and traveling screens) installed in the inlet line of the essential service water pumps would stop, since they do not have safety electrical supply (a rotating equipment having drum filter would continue to clean the water, and it has safety electrical supply). Each of the three screen systems requires regular cleaning. According to the operating experience the blockage process is slow; the significantly over-designed screens are capable to provide the essential service water pumps with sufficient quantity of clean water for several days. Nevertheless, it cannot be excluded that if the loss of the electrical power influences all four units, and thus the screen cleaning stops, then the drum filters will be blocked after a certain period of time, and then the ESWS will be lost. The same effect might be expected in a case of a seismic event, when the screens are damaged. The safety electrical power supply of band screens has to be solved in order to prevent the blockage of the screens of the essential service water system. The seismic resistance of the traveling screens needs to be enhanced to enable the functionality following a seismic event. Reference: HAEA (2011). National Report of Hungary on the Targeted Re-assessment of Paks Nuclear Power Plant, HAEA, December 2011. http://www.ensreg.eu/node/362

To be discussed	The issues of interest include: ➢ The design solution selected for the traveling screen safety related power supply system. ➢ Design solution for the seismic enhancement of the traveling screens. ➢ Are there any loads that need to be removed from the EDG backed busses to enable supply of traveling screens? ➢ Which consumers in the traveling screen installation are being supplied (e.g. main motor, heat tracing, backwash, etc.)? ➢ What is the total load to be added to the safety busses? ➢ As the SW intake is a single installation for the whole plant, to which unit(s) will the power supply be connected? What would be source of the power supply in a case of an outage of that unit?
Safety importance	High
Expected schedule	Medium term
Follow-up	Check list

HUNGARY	
Topic 2: Loss of safety systems	
Issue No	**HU 2.6**
Title	Reliable power supply of the water wells
Content	The electrical power supply for the submersible pumps of the water wells will be established from a (well-protected) fixed or mobile Diesel generator.
Safety relevance	As an alternate source of water (in a case the ESW is lost) Paks NPP made 9 water wells 30 meters deep, that are located at the banks of Danube. Each is equipped with a submersible pump and the pipework that connects to the ESW system. At present, as the pumps are powered by plants' normal power supply this option is not available to be used as ultimate heat sink if the external electric power supply is lost. Adding a dedicated DG to power those pumps will allow this alternate source to be used.
Background	One option for an alternative cooling water source is the application of the fire water systems of the plant. The essential service water system can be supplied from these sources, but currently they are capable to solve the alternative cooling water supply only in a limited extent. The fire water pump stations can be operated only if the normal electrical power supply is available; they have 2x2000 m^3 water inventory reserved in the discharge water canal; their permanent service can be guaranteed only if the cooling water systems are in operation. The fire water pump station operating with diesel pumps having fuel reserve that is sufficient for about eight hours of operation is also available., The current arrangement makes it possible to provide 100 m^3 water into the essential service water piping at the entry to the plant, from the (manually operated) connection between the fire protection system piping and the service water system piping In the case of total station black-out, the diesel pump station (that is independent of the electrical power supply grid) can be a valuable alternative source of cooling water; therefore the operator decided to introduce a corrective action in this regard in order to extend the water inventory to be used in the case of an accident. The nuclear power plant has 9 wells each having a large diameter and a depth of 30 m that are bored in the pebble bed of the Danube (this is the so called bank filtered well plant), these wells are permanent water sources providing practically unlimited quantity of water independently of the water level of the Danube. A connection system is installed from the well plant to the essential service water system. These wells will not be operable if the external electric power supply is lost, because 15 submersible pumps (385 kW) are supplied from the normal (non safety) electrical grid. Paks NPP developed a modification for provision of electrical power source independent of the external electrical power supply. In accordance with the NAcP, this arrangement shall be in place by 15. December 2013. Reference: HAEA (2011). National Report of Hungary on the Targeted Re-assessment of Paks Nuclear Power Plant, HAEA, December 2011. http://www.ensreg.eu/node/362

To be discussed	The presentation should describe in more detail the safety concept and design of the proposed measure and answer the following questions: ➢ What studies have been/are/will be made regarding the availability and capacity of the new source of water in extreme conditions (earthquake, drought, etc.) and what are the results (if already available)? ➢ What will be the external events resistance of the wells and associated equipment? ➢ What will be the configuration – safety train/cooling water source? ➢ From which sources (and which units) will the electrical supply be established?
Safety importance	High
Expected schedule	Medium term
Follow-up	Dedicated presentation

HUNGARY	
Topic 2: Loss of safety systems	
Issue No	**HU 2.7**
Title	External source for the make-up of spent fuel pool (SFP)
Content	The new cooling water supply pipe, resistant to external hazards, with fixed connection from the outside the building, will be installed to ensure the external cooling alternative to the SFP.
Safety relevance	At present, if the main SFP cooling/makeup system is lost, there will be no possibility to refill the SFP except from the inside. These areas might be subject to radiation and/or steam/heat due to evaporation of the water in SFP. Having a possibility of refilling the SFP by having a connection from outside a building is an important safety feature to be installed. The SFP could then be filled from the external borated water tanks, or other sources, (also a fire pump, given that the water is borated).
Background	The cooling of the spent fuel pool is performed by two independent redundant cooling circuits (each circuit is sufficient for the cooling of the SFP) containing a heat exchanger and a pump. The heat exchanger is cooled by the essential service water system. If the SW is lost, the cooling of the SFP is also lost. As a consequence the water in the SFP initially warms up and ultimately boils off, exposing the fuel elements leading to the damage of cladding. Based on conservative assumptions with the maximum inventory of fuel in the SFP the boiling process commences after about four hours, the damage to the cladding will commence after about 19 hours. Water in the SFP could be resupplied from the emergency core cooling system tanks. This action, for which an emergency procedure exists, would delay the heat up of the SFP, allowing for possible restoration of the SW. Nevertheless, without the SW, the SFP cannot be cooled, except by evaporation only in shutdown state, when the refuelling pool is filled up and connected to the RCS cooling could be performed by systems connected to the RCS, that being the RHR (which will work only if the SW is available) or by heat removal on the secondary side of SG (which might work even in the SW is lost). The refilling of the SFP could be accomplished by gravity feed from the upper trays of the localization tower. Nevertheless that water might be required for other purposes if a simultaneous accident occurs in the reactor. Manual operation of the valves in the flow path depends on the local radiological conditions (in some cases those valves might be inaccessible). For the sequences with permanent loss of the ultimate heat sink, Paks NPP plans to implement a corrective measure assuring the long term cooling of the spent fuel pools by the establishment of a new, independent and protected supply route. Water can be reasonably supplied by the use of mobile equipment, similar to the alternative supply to the containment, even from identical water sources. Since the issue of criticality may appear depending on the safety of the fuel stored in the spent fuel pool, thus the boron concentration of the water shall be set as of that supplied to the containment. In a severe accident, when water has to be provided to the containment, the establishment of the connections is influenced by the radiological conditions in the area. The access to the external connection point in accident conditions has

<table>
<tr><td></td><td>to be evaluated, and if required modified; this is planned to be implemented.
The water make-up to the SFP from an external source has to be made possible by the construction of a supply pipeline having adequate design against external hazards, with a connection from the yard. Water inventory with adequate boron concentration (see above) has to be supplied through this line to the SFP. The operating instructions on the practical application have to be developed. According to the NAcP, the deadline for the implementation is 15. December 2018.

Reference:
HAEA (2011). National Report of Hungary on the Targeted Re-assessment of Paks Nuclear Power Plant, HAEA, December 2011. http://www.ensreg.eu/node/362</td></tr>
<tr><td>To be discussed</td><td>The presentation should describe in more detail the safety concept and design of the proposed measure and answer the following questions:
➢ Proposed configuration, connection points.
➢ Source of borated water and control of boron concentration, and how to assure that non-borated water is not supplied to the SFP.
➢ Radiological conditions for access to connections in case of severe accident.
➢ Safety margin (e.g. time to fuel damage) gained by the implementation of the proposed solution.
➢ The schedule for the implementation, with specific emphasis as to why it is scheduled so far in the future.</td></tr>
<tr><td>Safety importance</td><td>High</td></tr>
<tr><td>Expected schedule</td><td>Short term</td></tr>
<tr><td>Follow-up</td><td>Dedicated presentation</td></tr>
</table>

<table>
<tr><td colspan="2">HUNGARY</td></tr>
<tr><td colspan="2">Topic 2: Loss of safety systems</td></tr>
<tr><td>Issue No</td><td>HU 2.8</td></tr>
<tr><td>Title</td><td>Additional water source for the diesel-driven fire water pumps</td></tr>
<tr><td>Content</td><td>Modifications will be implemented to enable additional 2x2000 m^3 water source from the closed section of the discharge water canal available for the suction of the diesel-driven fire water pump station.</td></tr>
<tr><td>Safety relevance</td><td>The fire water pump station of the Units 3&4, equipped with diesel-driven pumps, could, in a case when the essential service water system is non-operational, supply only 100 m^3 of water from its dedicated storage. In the case of total station black-out, the diesel pump station (that is independent of the electrical power supply grid) can be a valuable alternative source of cooling water, if the water supply would be assured. With this modification, a significant amount of water will be made available.</td></tr>
<tr><td>Background</td><td>The fire water system can be primarily considered as a water source. The primary water source of the fire water system is the bank filtered well plant, which is capable to provide 810 m^3/h water flow rate at a pressure of 8 bars. Additionally, the fire water pump station of the plant having 4000 m^3 water inventory is available, which is supplied from the discharge water canal of the units 1 and 2. The pumps of this station start automatically, should the need for fire water exceed the flow rate provided by the wells. If the pressure of the fire water system decreases below the lower service value, then the earthquake resistant diesel fire water pumps that suck from the outlet line of the essential service water system of Installation II start automatically.

The water base of the earthquake resistant fire water pump station of Installation II that is equipped with individual diesel power supply and capable to operate for eight hours can be utilized only if the cooling water systems are operating. The accessibility of the 2x2000 m^3 water reserve available in the closed segment of the discharge water canal has to be solved by implementation of necessary modifications for such cases when the supply from the essential service water system is lost. The deadline for this improvement, as stated in the NAcP is 15 December 2018.

Reference:
HAEA (2011). National Report of Hungary on the Targeted Re-assessment of Paks Nuclear Power Plant, HAEA, December 2011. http://www.ensreg.eu/node/362</td></tr>
<tr><td>To be discussed</td><td>The presentation should describe in more detail the safety concept and design of the proposed measure and answer the following questions:
➢ Engineering consideration, in particular in relation with the availability of the water reserve in a case of seismic event.
➢ Proposed configuration, connection points.
➢ Planned consumers to be supplied from the fire water pump station.
➢ Schedule for the implementation of this modification, and why it is in the NAcP planed relatively distant in the future.</td></tr>
</table>

Safety importance	High
Expected schedule	Short term
Follow-up	Dedicated presentation

HUNGARY	
Topic 2: Loss of safety systems	
Issue No	**HU 2.9**
Title	Extension of alternative cooling
Content	Additional connection points will be established on the demineralized water tanks to allow the water supply, through the auxiliary emergency feed water system, by mobile equipment. As water stored in the demineralized water tanks can be used as an alternative water source during an accident, appropriate connection points on these tanks will be used for providing water to the auxiliary emergency feed water system using mobile pumps.
Safety relevance	The demineralized water tanks play an important role in ensuring demineralized water stocks if the ultimate heat sink is lost, should it occur due to unmanageable low water level or any other reason. Three demineralized water tanks having 900 m^3 capacities each are installed per twin-units, which shall continuously include 500 m^3 water inventory as a minimum. In case of the accident when the normal in-house and grid electrical power supply are lost for 72 hours and the demineralized water preparation plant does not operate as well, the demineralized water tanks will be the only source of cooling from the secondary circuit. Even if the demineralized water preparation plant stops the make-up of the system, its water inventory can be used for long-term removal of the residual heat of the core.
Background	The function of the auxiliary emergency feedwater system is to supply water directly from the demineralised water tanks to make-up the steam generators to remove the residual heat of the reactor, should the normal feedwater and emergency feedwater systems fail. The delivery capacity of the pumps (two pumps per unit) is identical with the emergency feedwater pumps. The emergency feedwater system is totally independent of the auxiliary emergency feedwater system. If one of the systems fails, then the other will perform the safety function. The supply route of the auxiliary emergency feedwater system is independent of the normal service feedwater system. The water supplied by the system comes from the demineralised water tanks. The system, the demineralised water tanks and the connecting demineralised water pipelines are all seismic resistant. As a result of a previously implemented safety improvement measure the pumps and control valves of the system were moved from the turbine hall to the reactor building, where they are well protected from external effects. The system was then re-designed in compliance with the single failure criterion. The electrical power to the auxiliary emergency feedwater system is supplied from the Category II safety power supply, thus it starts automatically as supplied by the emergency diesel generators in the case of loss of the electrical power. According to the analyses, the water inventories are sufficient for three days to cool the units. When the demineralised water inventories run out, the alternative to supply water to the auxiliary emergency feedwater systems (those are independent for each unit) is through an independent external connection from another source. The mentioned connection to the outlet collectors of the auxiliary emergency feedwater systems of the twin units (that

	can be joined together) has been already installed. It should be mentioned that the auxiliary emergency feedwater pumps, the twin-unit connection possibility and the manual valves of the water supply from the yard are all installed in a single room. The pumps belonging to different units are physically separated in the common room by fire resistant walls. The safety implications of the installation in a common room, with its advantages and disadvantages, was previously evaluated in the probabilistic safety analysis of the nuclear power plant. An advantage of the common room is that it is enough to activate the mobile water supply at one point for both units of an installation. Water supply opportunities are available from the feedwater tanks through the existing filling pipelines of the high pressure preheaters; however, these sources are difficult to be supplied in the case of a severe accident. In order to facilitate the practical application during accidents, as an additional corrective measure, the installation of points at the demineralized water tanks where mobile connections are possible were envisaged. The NAcP indicates the 15. December 2014 as the deadline for the completion of this improvement. Reference: HAEA (2011). National Report of Hungary on the Targeted Re-assessment of Paks Nuclear Power Plant, HAEA, December 2011. http://www.ensreg.eu/node/362
To be discussed	➢ The presentation should describe in more detail the safety concept and design of the proposed measure, including the design basis, the analysis undertaken or planed, the issues related with the implementation, equipment to be installed and procedures to be modified. Also of interest are the licencing issues, if any.
Safety importance	High
Expected schedule	Medium term
Follow-up	Dedicated presentation

3.3 Topic 3: Severe Accident Management

Czech Republic / Hungary / Slovakia	
Topic 3: Severe Accident Management	
Issue No	**CZ/HU/SK 3.1**
Title	Stabilization of molten core for reactors of the type VVER-440/213 (Bohunice, Dukovany, Mochovce, Paks)
Content	Implementation of this measure – stabilization of the molten core by cooling the reactor pressure vessel from outside – was already planned before the Fukushima accident, and indeed was already completed at some units at the time of the accident. The measure requires a number of technical modifications. Since the cooling of the RPV from the outside is a complex procedure, extensive analyses and experiments have been performed to demonstrate the feasibility. Of particular importance is the CERES test facility which permits to simulate the gap between RPV and biological shield 1:1 regarding elevation, with a 1:40 slide of the cylindrical structure. Furthermore, considerations for the case of failure of this measure have been performed in the three countries concerned. The assessment of and the approach to this problem appears to differ between the countries.
Safety relevance	There are two options to attempt to stabilize a molten core: Inside the reactor pressure vessel, by external vessel cooling; or, after melt-through of the RPV, by cooling in the reactor cavity. For smaller reactors, in particular VVER-440s, the former option (in-vessel retention) could, in principle, be practicable. (For larger reactors – roughly above 1.000 MWe – in-vessel retention does not appear feasible due to a less favourable ratio between decay heat and RPV surface.) Successful in-vessel retention leads to rather limited pressure increase in the containment (for VVER-440s, this is supported by the relatively large volume of the containment), and to limited release of radionuclides into the containment atmosphere. Comparatively low releases into the environment are the result. Insofar, the implementation of filtered venting can be seen with less urgency for VVER-440/213 than for VVER-1000. Without cooling and stabilization of the molten core inside the reactor vessel, containment failure appears likely. There appear to be differences in the assessments regarding the possible accident sequences in this case, and the severity of resulting releases, in the countries discussed here; the basis for these differences is not clear, and this point should be pursued further.
Background	<u>Implementation of external reactor pressure vessel (RPV) cooling</u> A number of technical modifications have to be performed to implement external cooling of the RPV: Modification of the drainage system of the bubble condenser, modifications in the reactor shaft to permit coolant flow along the RPV, modification of the ventilation piping to avoid losses of cooling water, strengthening of the hermetic door of the reactor cavity and others. According to the **Peer Review Country Reports** (ENSREG 2012a, 2012b, 2012c) and other sources, the schedule for implementation is as follows:

	EDU – until 2015 Paks – between 2011 (unit 1) and 2014 (unit 4) EBO – 2010 EMO 1+2 – 2011/12 (EMO 3+4 – part of the original design) Thus, the implementation is already quite far advanced and it can be expected to continue according to the planned schedule. <u>Demonstration of feasibility of external RPV cooling</u> It is generally assumed (by the licensees as well as, subject to further review, the regulatory authorities) that the risk of vessel failure can be significantly reduced by implementing the strategy of cooling the reactor pressure vessel from outside. Analyses have been performed to investigate whether stable cooling can be assured through natural circulation of the coolant, maintaining the intactness of the RPV. In support of the calculations, experiments have been performed in the CERES test facility in Hungary. Information on analyses and experiments have been provided by the Hungarian side at the regular bilateral meeting Hungary-Austria 2012: 1. Research Results in Support of In-vessel Corium Retention Program in the Paks Nuclear Power Plant (lecture at European Review Meeting on Severe Accident Research (ERMSAR) 2012) 2. CERES experiments calculation with the ASTEC code (lecture at ERMSAR 2012) 3. CERES test facility and test results (presentation at regular bilateral meeting Hungary-Austria 2012) The first paper describes the CERES test facility which simulates the gap between RPV and biological shield (1:1 regarding elevation, with a 1:40 slide of the cylindrical structure). Results of experiments for different gap configurations are presented, as well as results of calculations for one case. It is concluded that removal of the decay heat could be demonstrated in all cases. The second paper provides results of analyses for another gap configuration. It concluded that there is good agreement between experiment and calculations, and that the coolability of the RPV has been demonstrated. The third document mostly summarizes the other two. The CERES experiments were mostly completed in late 2012. There was one remaining issue at that time: A test with boric acid, which was planned for 2013. No information on other comparable investigations has come to the attention of the Austrian experts. It can be assumed that the CERES experiments and the calculations carried out in this context constitute the mainstay of the demonstration of feasibility of external RPV cooling. <u>Considerations for the case of RPV failure</u> Different considerations regarding RPV failure have been performed in the three countries concerned. In the **Czech Republic**, the emphasis lies on cooling the steel door of the reactor shaft by flooding the shaft. No analysis has been performed; but according to

"professional estimate", failure of the door can be prevented. This would be followed by melt-through of the wall of the shaft after about 4 days after failure of the RPV bottom. It is pointed out that this *represents high and late damage to the containment. The concentration of fission products in the atmosphere of the containment would be low at this time* (National Stresstest Report (SÚJB 2011) section II.6.2.3, repeated in the Czech Report to the 2nd CNS EOM (CR 2012)). No information is available whether further analyses and preparation of measures is planned in this respect.

In **Hungary**, two cases are distinguished: RPV failure before flooding of the reactor cavity, and after it. In the first case, it has to be decided whether flooding of the cavity should be still be performed, taking into account the possibility of a steam explosion. In the second case, *a relatively small amount of molten fuel will escape and then the solidifying debris will block the route* (National Stresstest Report section (HAEA 2011a) 6.2.3). This seems to imply that RPV failure does not lead to major problems as long as flooding occurs sufficiently early. The basis for this statement is not clear; no information is provided whether there are analyses supporting it, or whether further analyses are planned.

In **Slovakia**, it is assumed that failure of the cavity door is unlikely to be prevented in case of RPV failure. The failed door is expected to lead to releases outside the containment and a serious worsening of the accident progression. *Stabilization of the melt composition, termination of concrete degradation and long-term preservation of the cavity integrity cannot be guaranteed* by coolant feeding into the reactor cavity. Therefore, RPV failure prevention is given high importance and *no special additional measures were assumed for hypothetical corium cooling on the cavity bottom* (National Stresstest Report (UJDSR 2011) 6.3.5.2). The Slovak Report to the 2nd CNS Extraordinary Meeting (SR 2012) contains similar statements.

In the Slovak National Action Plan (NAcP) (UJDSR 2012), this point is again emphasized: *Implementation of reliable in-vessel molten corium retention prevents complicated ex-vessel phenomena associated with core-concrete interaction, direct containment heating, production of non-condensable gases leading to containment over pressurization, etc.; all these phenomena are associated with large uncertainties (part III, section 'severe accident management')*.

It is noteworthy that in the Peer Review Country Report (ENSREG 2012b), it is stated that RPV failure is considered very unlikely after the modifications for in-vessel retention. *Nevertheless, investigation to limit the consequences in case of RPV failure could be considered in further steps* (section 4.3).

References:

CR (2012). Czech Republic – Extraordinary National Report under the Convention of Nuclear Safety, February 2012. http://www.sujb.cz/fileadmin/sujb/docs/zpravy/narodni_zpravy/CZ_NR_2012.pdf

ENSREG (2012a). Peer review country report – Czech Republic. Stress tests performed on European nuclear power plants. http://www.ensreg.eu/node/393

ENSREG (2012b). Peer review country report - Slovakia. Stress tests performed on European nuclear power plants.

	http://www.ensreg.eu/node/404 ENSREG (2012c). Peer review country report – Hungary. Stress tests performed on European nuclear power plants. http://www.ensreg.eu/node/398 HAEA (Hungarian Atomic Energy Authority) (2011a). National Report of Hungary on the Targeted Safety Re-assessment of Paks Nuclear Power Plant, December 29, 2011. http://www.ensreg.eu/node/362 SR (2012). Special National Report of the Slovak Republic, compiled under the Convention of Nuclear Safety, April 2012. http://www.ujd.gov.sk/files/SNR_NS_April2012.pdf SÚJB (2011). National Report on "Stress Tests" NPP Dukovany and NPP Temelin Czech Republic. Evaluation of Safety and Safety Margins in the Light of the Accident of the NPP Fukushima. http://www.ensreg.eu/node/369 UJDSR (Nuclear Regulatory Authority of the Slovak Republic) (2011). The Stress Tests for Nuclear Power Plants in Slovakia. 30. December 2011. http://www.ensreg.eu/node/366 UJDSR (2012). Post Fukushima National Action Plan (NAcP) of the Slovak Republic. http://www.ensreg.eu/node/692
To be discussed	This measure - stabilization of the molten core by cooling the reactor pressure vessel from outside - has already been decided, the corresponding modifications have been planned in detail, and the implementation is already far advanced (by the end of 2013, it will be completed in more than half of the units concerned), although it follows different schedules in the different countries. The discussion should therefore focus primarily on the demonstration of the feasibility, and also on the considerations for the case of failure of the measure. <u>Demonstration of feasibility</u> The information provided by the Hungarian side (see above) gives an overview of the programme performed in Hungary to demonstrate the feasibility of in-vessel retention. The CERES test facility follows the geometry at Paks NPP. There may be some small differences in geometry between the VVER-440/213s under consideration here, but it can be assumed that the CERES results are also important for the other plants. After evaluation of the information provided, a number of questions remain open: ➢ Has the test with boric acid, planned for 2013, already been performed? If so, what are the results? ➢ The experiments are modelling a part of the whole system only (the cooling of the external vessel wall). The overall concept (e.g. containment spray system, piping from sump to reactor cavity) should be described in more detail. ➢ Two load cases have been calculated with ASTEC/ANSYS. It is not clear to which extent they are representative for the whole spectrum of accidents.

	➢ Different widths of the gap between RPV and cavity wall have been studied in experiments and calculations. However, the case of complete local gap closure was not considered, as far as can be seen. Can this case be excluded? If not, what would be the effect of a local closure? ➢ In the tests, stepwise increase of the thermal power has been implemented. It is not clear that all relevant cases are covered. ➢ The experiments show, that boiling crisis, drying-out of the wall and local temperature increases to up to 200° above boiling temperature can occur for brief periods of time. Subsequently, the wall is cooled again to boiling point when water flows up again. Have structure-mechanical analyses been performed to study possible consequences of this heating-cooling cycle of the RPV wall? ➢ The codes used for calculations (RELAP5 and ASTEC) predict the mass flow well; however, both codes appear to have difficulties in correctly predicting the boiling crisis at the wall. ➢ How reliable is the transfer of the results from a 1:40 slide to the full RPV circumference? Reliable codes are needed for such a transfer. Are RELAP5 and ASTEC adequate for this task, considering their limitations in predicting experimental results? ➢ Are there differences in geometry and/or other differences regarding the whole concept of IVR, between Paks and the other VVER-440/213s considered here? If so, what are the differences and how can the results of CERES be transferred to other plants in spite of these differences? <u>Considerations for the case of RPV failure</u> Different considerations have been performed in different countries. All in all, there is a number of questions which appear relevant: • When the cavity is flooded after RPV failure, there is the hazard of a steam explosion. Should flooding be avoided completely in this case, or could there be circumstances in which it might be advantageous nevertheless? Are further analyses and investigations planned in this respect? • What is the basis for the assumption that only a relatively small amount of molten fuel will escape and then the route will be blocked by solidifying debris (as assumed in Hungary)? Are further analyses and investigations planned in this respect? • What is the basis for assuming that the integrity of the cavity door can be preserved by flooding (Czech Republic)? Further analyses and investigations planned? • What is the basis for assuming that melt-through of the shaft will occur after about 4 days (Czech Republic)? To which extent will releases from the containment be reduced in this case, compared to early containment failure through failure of the cavity door? Which further analyses and investigations are planned? The CERES experiments were expected to be completed by the end of 2013, and it can be assumed that the considerations for the case of RPV failure are on-going. The appropriate time for a workshop could be early 2016.

Safety importance	High
Expected schedule	Medium term
Follow-up	Dedicated workshop

HUNGARY	
Topic 3: Severe Accident Management	
Issue No	**HU 3.2**
Title	Avoid long-term over-pressurization of containment
Content	After flooding of the reactor cavity, the evaporation of coolant will gradually increase pressure in the containment, if sprinklers are not available. If there is no pressure reduction, the containment might fail due to over-pressurization after several days. To avoid this, either filtered venting or additional measures to assure long-term containment cooling by the containment sprinklers could be introduced. The latter option has been selected for Paks NPP. The implementation of this measure requires extensive designing and installing work. The deadline is end of 2018. It must be noted that this measure can only be adequate in case of successful in-vessel retention of the molten core.
Safety relevance	If in-vessel retention of the molten core is successful, pressure increase in the containment is limited and only a relatively small amount of non-condensable gases is produced. Therefore, filtered venting has not been envisaged so far for VVER-440/213s. However, late containment failure due to over-pressure can occur even with successful in-vessel retention. In this case there will be radioactive releases which, although significantly smaller than releases in case of early containment failure, are still radiologically significant. It can be regarded as a good practice that the possibility of long-term over-pressurization has been analysed in Hungary, and that measures are under consideration.
Background	This issue is dealt with in the **National Stresstest Report** (HAEA 2011). It is stated that after flooding of the reactor cavity in case of a severe accident, the evaporation of coolant will gradually increase pressure in the containment, if the sprinkler system is not available. (It is assumed that the flooding of the reactor cavity is successful and the molten core is stabilized inside the reactor pressure vessel.) Analyses of the long-term development have been carried out. The HCLPF value for the containment pressure (3.35 bar absolute) will be reached within 3 to 8 days, depending on the containment leakage rate. If pressure reduction does not occur, pressure will increase until the containment fails or until the mass flow of the leakage will equal the generated mass flow (section 6.3.3). The bubble condenser system is not available in this situation, since water from the bubble condenser trays is used for flooding the reactor cavity (see Issue CZ/HU/SK 3.1). Information concerning the containment leakage rate of the 4 units at Paks has been made available in the course of the **Paks Roadmap** (UBA 2012, section 6). SAMGs require the reduction of the pressure in the containment. This can be achieved by cooling of the containment atmosphere with the sprinkler system, or by unfiltered venting. In case of total loss of electricity supply, cooling is not

	available. An unfiltered release can only be executed after the evacuation of the area around the NPP. Therefore, further measures were reported to be under consideration; either the introduction of filtered venting, or additional measures to assure long-term cooling of the containment by the sprinklers (HAEA 2011a, section 6.3.3). In the Stresstest **Peer Review Country Report** (ENSREG, 2012c) for Hungary, this issue is discussed in the section "weak points, deficiencies" (4.2.2.2). The two alternatives are mentioned again. In section 4.3, among other issues, study of measures to prevent over-pressurization of the containment is recommended. It is pointed out that these measures are considered to be adequate only in case of successful in-vessel retention of the molten core (see Issue No HU 3.1). The **National Action Plan** (HAEA 2012a) states that a system to prevent long-term, slow over-pressurization of the containment shall be developed and implemented. Furthermore, it is commented that Paks NPP prepared the concept for implementation, recommending the installation of an active containment cooling system. Final deadline is 15.12.2018 (Part IV, Task 30). The deadline (2018) is comparatively late (given the importance of the issue). The reason possibly is that extensive work is required for designing and installing this measure. At the **Bilateral Meeting 2013**, the Hungarian side confirmed that the method selected for containment overpressure protection is the installation of an active system for containment cooling, to enable the use of the containment spray in case of severe accidents. The system is to be equipped with condensers and a dry cooling tower (BM A-HU 2013). References: BM A-HU 2013. Presentation "Status of the National Action Plan at the Paks NPP in Hungary" at the 19th Hungarian-Austrian Bilateral Meeting, Budapest, November 18/19, 2013 ENSREG (2012c). Peer review country report – Hungary. Stress tests performed on European nuclear power plants. http://www.ensreg.eu/node/398 HAEA (Hungarian Atomic Energy Authority) (2011a). National Report of Hungary on the Targeted Safety Re-assessment of Paks Nuclear Power Plant, December 29, 2011. http://www.ensreg.eu/node/362 HAEA (2012a). National Action Plan of Hungary on the implementation actions Decided upon lessons learned from the Fukushima Daiichi accident. http://www.ensreg.eu/node/685 UBA (2012). Roadmap of Paks NPP Lifetime Extension (LTE) – Summary of Current Status and Open Questions of the Bilateral Meetings between Austria and Hungary. Rep-0402. http://www.umweltbundesamt.at/aktuell/publikationen/publikationssuche/publikationsdetail/?pub_id=2015

To be discussed	Questions which should be addressed in a presentation are: ➢ Information on the containment leakage rate as measured in a full pressure test, at units 1, 2 and 4. ➢ Which considerations led to the preference of containment cooling vie the sprinklers, rather than filtered venting? ➢ Brief description of the system which is to be installed, including the p ower supply. ➢ Current status of work and schedule for completion. ➢ How will safety be improved by this measure? How does the state of the NPPs before implementation compare with the state after implementation of the measure? This issue can be discussed as soon as the planning for the new system has been completed, which can be expected soon.
Safety importance	High
Expected schedule	Medium term
Follow-up	Dedicated presentation

HUNGARY	
Topic 3: Severe Accident Management	
Issue No	**HU 3.3**
Title	Study of hydrogen generation and distribution in the reactor hall
Content	Large amounts of hydrogen can be produced in case of a severe accident with simultaneous damage in two reactors and/or two spent fuel pools. The hydrogen could be released into the atmosphere of the reactor hall of a twin unit. Analyses were to be performed with three-dimensional models to determine the quantity and distribution of hydrogen in the reactor hall. Furthermore, a new coolant supply route to the SFPs will be installed, to prevent or at least delay hydrogen generation in the pools.
Safety relevance	In case of an accident with simultaneous damage to more than one reactor and/or SFP, significantly larger amounts of hydrogen can be produced than in case of a more limited accident, leading to higher hydrogen hazard and the need for more extensive counter-measures. Deflagration concentrations may be reached in the reactor hall. Other countries with VVER-440s (in particular, the Czech Republic and Slovakia) did not announce that such studies are planned. Study of the hydrogen issue in case of an accident with several hydrogen sources in the reactor hall can be regarded as good practice.
Background	The regulatory authority mentioned this issue for further study in the **National Stresstest Report** (HAEA 2011a). Analyses to determine the quantity and distribution of hydrogen in the reactor hall are to be carried out by the licensee, by order of the regulator. They are to be based on the assumption of an accident with two damaged SFPs and two damaged reactors within a twin unit. The analyses shall be performed with three-dimensional models rather than lumped-parameter models (section 7.3.1). This is also mentioned in the Stresstest **Peer Review Country Report** (ENSREG 2012c, section 4.3). The **National Action Plan (NAcP;** HAEA 2012a) states that three-dimensional analyses going beyond the use of lumped-parameter models (being more accurate and less conservative) shall be performed. The results of these analyses will show whether inflammable concentration might occur, leading to turbulent burning. The need for further action will be determined based on these results. Final deadline is 31.12.2012 (Part IV, Task 41). Results have not been published so far. A new coolant supply route to the SFPs is to ensure the possibility to prevent or at least delay the generation of large volumes of hydrogen in the pools. References: ENSREG (2012c). Peer review country report – Hungary. Stress tests performed on European nuclear power plants. http://www.ensreg.eu/node/398 HAEA (Hungarian Atomic Energy Authority) (2011a). National Report of

	Hungary on the Targeted Safety Re-assessment of Paks Nuclear Power Plant, December 29, 2011. http://www.ensreg.eu/node/362 HAEA (2012a). National Action Plan of Hungary on the implementation actions Decided upon lessons learned from the Fukushima Daiichi accident. http://www.ensreg.eu/node/685
To be discussed	Questions which should be addressed in a presentation are: ➢ Description of the analyses which were performed (details of accident scenarios considered, methods, results). ➢ Description of the measure to prevent or delay hydrogen releases from SFPs (technical features, schedule for implementation). ➢ How will safety be improved by this measure? How does the state of the NPPs before implementation compare with the state after implementation of the measure? The analyses have already been completed and planning of the new measure appears to have at least started. A presentation could take place in the short term.
Safety importance	High
Expected schedule	Short term
Follow-up	Dedicated presentation

HUNGARY	
Topic 3: Severe Accident Management	
Issue No	**HU 3.4**
Title	Measures against containment bypass via steam generator
Content	Bypass of the containment can lead to high releases in case of accidents. Steam generator tube or collector rupture constitutes one important case of containment bypass. Blowdown lines at the secondary side of the SGs have been installed at Paks for returning the leaked coolant into the containment. This measure has already been implemented, but no detailed information has been available so far.
Safety relevance	Accidents with the steam generator (SG) tube or collector rupture can lead to particularly high releases, since the containment is bypassed. Furthermore, primary coolant can leave the containment in this case, and will not be available for recirculation from the containment sump.
Background	The issue of containment bypass by steam generator tube or collector rupture has been noted as being of importance by the Austrian side in 2009 but has not been discussed at the **Paks Roadmap** sessions due to lack of time. Elter et al. (2009) state that PSA results indicate, that the risk of large releases is dominated by containment bypass sequences. The implementation of blow-down lines which are directed to the containment, at the secondary side on the bottom of the steam generators, is mentioned as an effective precaution. This measure is listed among "possible accident management measures". According to a HAEA publication of October 2011, the implementation of this measure was completed in 2011 with the installation of new valves (plug valves) in all four units (HAEA 2011b). This measure is not mentioned in the NAcP (HAEA 2012a). Although this measure has already been completely implemented, more detailed information would be of high interest, in particular in the context of other SAM measures implemented or planned post-Fukushima. References: Elter, J., et al. (2009). Development of the SAM strategy for Paks NPP on the basis of Level 2 PSA. Workshop Proceedings of ISAMM 2009: Implementation of Severe Accident Management Measures. Schloss Böttstein, Switzerland, October 26-28, 2009 HAEA (2011b). Recent Developments in Nuclear Safety in Hungary, October 2011. http://www.haea.gov.hu/web/v2/portal.nsf/download_en/4A5AF954DAF288B7C12579F7003AECD3/$file/Recent%20Developments%202011_2.pdf HAEA (2012). National Action Plan of Hungary on the implementation actions Decided upon lessons learned from the Fukushima Daiichi accident. http://www.ensreg.eu/node/685

To be discussed	Questions which should be addressed in a presentation are: ➢ Detailed description of the measure. ➢ Have there been any problems due to interaction with other SAM measures post-Fukushima? ➢ How was safety improved by this measure? How does the state of the NPPs before implementation compare with the state after implementation of the measure? Since implementation of this measure is already completed for some time, a presentation could take place in the short term.
Safety importance	High
Expected schedule	Short term
Follow-up	Dedicated presentation

HUNGARY	
Topic 3: Severe Accident Management	
Issue No	**HU 3.5**
Title	SAMGs to manage multi-unit accidents and simultaneous accidents in reactor and SFP
Content	As the Fukushima accident has illustrate, accidents in reactor and SFP can be initiated simultaneously in case of severe external events, potentially leading to significant releases. It is important to have SAMGs developed and implemented for such situations. SAMGs developed and implemented so far at Paks do not take multi-unit accidents into account. Regarding simultaneous management of SFP and reactor core, systems are reported to be available, but guidelines for use of resources need to be developed. Corresponding measures are foreseen to be implemented by 2017/18.
Safety relevance	In case of severe external events, accidents in reactor and SFP can be initiated simultaneously, potentially leading to significant releases. It is important to have SAMGs developed and implemented for such situations.
Background	In the **National Stresstest Report** (HAEA 2011a, section 6.3.8), it is pointed out that SAMGs refer to the use of systems of the twin-unit, which is clearly not possible in case of a multi-unit accident. Also, the simultaneous accident management in more than one unit means increases organizational tasks for the personnel. On the other hand, resources and electrical energy supply required for accident management are installed individually in each unit; from this point of view, the management of multi-unit accidents is solved. It is also stated that systems required for the simultaneous management SFP and reactor core are available; however, the guideline on the use of resources is not yet prepared and has to be developed (section 6.3.9). In the **Peer Review Country Report** for Hungary (ENSREG 2012c), it is stated that HAEA requested the development of SAMGs for simultaneous accidents in SFP and reactor, and also an analysis of the resources needed for the on-site management of multi-unit accidents (section 4.2.4.2). These points are also supported by the peer reviewers (section 4.3). The **National Action Plan** (HAEA 2012a) contains several actions relevant in this context in Part IV: <u>Task 34:</u> *Severe accident situations simultaneously taking place in the reactor and the spent fuel pool shall be managed by the development of a severe accident management guideline.* It is commented in the NAcP that the guidelines will enter into force between the end of 2012 and 2014, when the technical modifications in the different units are completed. However, 15.12.2018 is given as final deadline. <u>Task 36:</u> Extension of the capabilities of the Technical Support Center to provide sufficient resources for simultaneous management of severe accidents in up to

	all 4 units, to be completed 15.12.2018. Task 37: Development of procedures of personnel and equipment provisions in case of multi-unit accidents, to be completed 15.12.2017. References: ENSREG (2012c). Peer review country report – Hungary. Stress tests performed on European nuclear power plants. http://www.ensreg.eu/node/398 HAEA (Hungarian Atomic Energy Authority) (2011a). National Report of Hungary on the Targeted Safety Re-assessment of Paks Nuclear Power Plant, December 29, 2011. http://www.ensreg.eu/node/362 HAEA (2012). National Action Plan of Hungary on the implementation actions Decided upon lessons learned from the Fukushima Daiichi accident. http://www.ensreg.eu/node/685
To be discussed	Questions which should be addressed in a presentation are: ➢ Which changes and new procedures have been introduced in the first phase (until 2014)? How was safety improved by these measures? ➢ What is planned for the second phase (until 2018)? What is the detailed schedule? How will safety be improved by the new measures, compared to the status before? An appropriate time for a presentation could after completion of the first phase, with some buffer time allowing for delays.
Safety importance	High
Expected schedule	Medium term
Follow-up	Dedicated presentation

HUNGARY	
Topic 3: Severe Accident Management	
Issue No	**HU 3.6**
Title	Severe accident scenarios / PSA
Content	Level 1 and 2 PSAs have been performed for Paks pre-Fukushima, and some results have already been provided. Currently, a new level 2 PSA is on-going, taking into account the impact of new SAM measures. Results for power operation should be available 2013, for shutdown 2014/15. There are a few open questions from the Paks Roadmap discussions, concerning the effect of a recent power uprate on severe accident scenarios (intervention times, hydrogen production etc). These could be discussed in the context of the PSAs, which necessarily involve the study of severe accidents.
Safety relevance	The significance of the overall results of PSAs (in particular, CDF and LRF) is rather limited, due to a number of factors which are inherent to PSAs and lead to considerable uncertainties in the results. Nevertheless, a PSA is a very useful tool to identify vulnerabilities in an NPP, as an important input for deciding on backfitting measures. It can also be helpful, although with high uncertainties, to quantify releases. Regarding the power uprate, there is a potential for accelerating accident sequences and increasing releases in case of severe accidents. It is of high interest to ascertain that appropriate measures have been taken to counteract these effects, to neutralize them or to render them insignificant. The two topics of PSA level 2 and PU (as far as potentially relevant for severe accidents) are connected and hence are treated here in a single Issue.
Background	At the **Paks Roadmap** session of the Bilateral Meeting in **2009**, background information on frequencies of different release categories and on source terms was provided (UBA 2012, sections 5 and 6). Also, the influence of hydro accumulator pressure reduction (as measure in the context of power uprate) on the development of accident sequences was discussed in a general manner. The measures which already have been implemented or are planned as result of the EU stresstest will influence severe accident source terms, and large release frequencies. Thus, an update for this issue, as soon as comprehensive and significant new results are available, would be appropriate. At the **Bilateral Meeting 2012**, the Hungarian side stated that a level 2 PSA study is at present being performed, taking into account impact of new SAM measures. Results for power operation should be available 2013, results for shutdown, open reactor and SFP 2014/15 (BM A-HU 2012). The **National Action Plan** (HAEA 2012a) only states that Level 1 and 2 PSAs have been performed for Paks, for each operating mode and the SFPs. Low probability event sequences are not excluded from the consideration of SAM, based on PSA results, since the SAMGs are not event based, but symptom based. It is stated that no action is seen as necessary (Part IV, 3.1.15). In this context, some questions from the **Paks Roadmap** discussions (UBA 2012, section 5) concerning changes in accident scenarios due to the power uprate (PU) which have remained open are also of interest. In the course of the Roadmap, many questions have been clarified and significant information has

	been made available. However, some points were discussed only generally and summarily and more detailed information would be of interest: a) In case of SBLOCA without successful HP injection, how long is the time until overheating after PU and for the previous power level? Does the power uprate make a relevant difference in this respect? b) Is there quantitative information on available time frames for successful operator actions for other cases (for both power levels)? c) During Paks Roadmap discussions, it has been stated that the integrated mass of H2 produced after PU is higher than before, but the production rate is lower. Is there quantitative information on the changes in the integrated mass of produced H_2 and the production rate after PU? d) During Paks Roadmap discussions, it has been stated that proposed mitigative SAM actions remain essentially unchanged after the PU, and that only some small modifications were required. Can more information on the evaluation of SAM mitigative actions in relation to PU, and the required small modifications, be provided? The consequences of the power uprate which was performed at Paks are not subject of the National Action Plan. References: BM A-HU (2012). Presentation "SAM strategy and plant modifications at Paks NPP" at the 18th Hungarian-Austrian Bilateral Meeting, Eisenstadt, December 07, 2012. HAEA (2012). National Action Plan of Hungary on the implementation actions Decided upon lessons learned from the Fukushima Daiichi accident. http://www.ensreg.eu/node/685 UBA (2012). Roadmap of Paks NPP Lifetime Extension (LTE) – Summary of Current Status and Open Questions of the Bilateral Meetings between Austria and Hungary. Rep-0402. http://www.umweltbundesamt.at/aktuell/publikationen/publikationssuche/publikationsdetail/?pub_id=2015
To be discussed	Questions which should be addressed in a presentation are, taking into account information already received by Austrian side during Roadmap discussions: ➢ Information about the new level 2 PSA study which takes into account the new SAM measures (scope, methods, results – in particular, source terms and large release frequencies). Which main improvements do the results of the level 2 PSA show? Open questions from the Paks Roadmap discussions which are relevant in the context of severe accidents: ➢ In case of SBLOCA without successful HP injection, how long is the time until overheating after PU and for the previous power level? Does the power uprate make a relevant difference in this respect? ➢ Is there quantitative information on available time frames for successful operator actions for other cases (for both power levels)? ➢ Is there quantitative information on the changes in the integrated mass of produced H_2 and the production rate after PU?

	➢ Can more information on the evaluation of SAM mitigative actions in relation to PU, and the required small modifications, be provided? An appropriate time would be after completion of the new level 2 PSA, which includes the parts for shutdown, open reactor and SFPs, with some buffer time allowing for delays.
Safety importance	Medium
Expected schedule	Long term
Follow-up	Dedicated presentation

3.4 Topic X: Outside Topics 1 - 3

Hungary	
Outside topics 1 - 3	
Issue No	**HU X.1**
Title	Reactor pressure vessel integrity
Content	Demonstrating the integrity of the reactor pressure vessel over the plant lifetime is a complicated task. Material properties and their development over time, thermo-hydraulic accident analyses, fracture-mechanic analyses and information on the accuracy of inspection techniques are required. In the discussions of the **Paks Roadmap** procedure (UBA 2012), this Issue was extensively discussed and significant information provided by the Hungarian side. Most of the questions of the Austrian experts were answered. However, a few points have not been cleared so far. The open points concern material properties, safety margins as derived in PTS-analyses, dose rate effects and the surveillance program.
Safety relevance	Guaranteeing the integrity of the reactor pressure vessel is of foremost importance since in case of vessel failure, core cooling cannot be provided by safety systems and a severe accident with high radioactive releases is likely.
Background	Concerning the database for un-irradiated material, it was briefly mentioned in a Hungarian document provided 2008 (Fekete & Toth 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 the document of 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 (whereas the dose rate effect related to thermal ageing has been discussed). It was stated in the document of 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.

	References: Fekete & Toth (2008). Fekete, T. & Tóth, P.: Summary of PTS Calculations. Budapest, 2008. Provided for the 14th bilateral meeting in 2008 by the Hungarian side. UBA (2012). Roadmap of Paks NPP Lifetime Extension (LTE) – Summary of Current Status and Open Questions of the Bilateral Meetings between Austria and Hungary. Rep-0402. http://www.umweltbundesamt.at/aktuell/publikationen/publikationssuche/publikationsdetail/?pub_id=2015
To be discussed	The following questions should be addressed in a presentation: ➢ Regarding the database of un-irradiated material, what is the scope of the recording of properties? ➢ Does the difference of 25 °C (see above) 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)? ➢ Information on the method for monitoring operational changes by using specimens of the surveillance programme. ➢ Is there new information from the surveillance program? A significant amount of information should be available now. However, it might be appropriate to discuss this issue at a later date when new information might have become available.
Safety importance	Medium
Expected schedule	Medium term
Follow-up	Dedicated presentation

Hungary	
Outside topics 1 - 3	
Issue No	**HU X.2**
Title	Ageing management
Content	A comprehensive system of ageing management is required for an NPP, particularly in case of lifetime extension. Hence, this Issue was discussed extensively in the framework of the **Paks Roadmap** procedure (UBA 2012). Significant information provided by the Hungarian side. Many questions of the Austrian experts were answered. However, a few points have not been cleared so far. They concern the experience with the ageing management program so far, and approach for the adoption of the ASME code at Paks NPP.
Safety relevance	Ageing management is important, in particular in the context of lifetime extension, to guarantee that any deterioration of systems, structures and components which might be relevant for safety is discovered in a timely manner, and appropriate counter-measures can be taken. Minor events and irregularities with little safety significance by themselves can be important as indicators for potentially safety-relevant problems.
Background	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 a Hungarian document (Pinczes 2009), full implementation of the AMP should be accomplished today. The available information shows that a comprehensive and systematic approach for ageing management has been implemented in Paks NPP – at least for mechanical components (no information concerning ageing management of structures and I&C components has been made available). The database/expert system DACAAM is an integral part of the AMP, which appears 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. Relevant in this context 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 (e.g. leaks of a steam generator drainage pipe due to corrosion or erosion, and a water purification system pipe in unit 4 presumably caused by thermal fatigue (HAEA 2011c)). According to a report of the regulatory authority (HAEA 2012b), the Hungarian licensing procedure for the life extension is similar to the U.S. NRC approach in license renewal according to 10 CFR 54. The new Hungarian regulations 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 requirements of ASME Code XI. 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 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. References: HAEA (2011c). Recent Developments in Nuclear Safety in Hungary, April 2011. http://www.oah.hu/web/v2/portal.nsf/download_en/42A7926CF3D243BFC1257881002EC1B5/$file/Recent%20Developments%202011_1.pdf HAEA (2012b). Recent Developments in Nuclear Safety in Hungary, November 2012. Download from www.oah.hu on December 05, 2012; currently not available on HAEA website. Pinczes (2009). Pinczes, J.: Development of aging management and in-service inspection. Budapest, 2009. Provided for the 15th bilateral meeting in 2009 by the Hungarian side.
To be discussed	The following questions should be addressed in a presentation: ➢ 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, taking into account to efficiency of cooperation and coordination of different departments as well as recent leakage events. ➢ 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. A significant amount of information should be available now. However, it might be appropriate to discuss this issue at a later date when more information about experiences with the AMP might have become available.
Safety importance	Medium
Expected schedule	Long term
Follow-up	Dedicated presentation