Post-
Fukushima
accident

Hungary

Peer review country report

Stress tests performed on European nuclear power plants

ENSREG
European Nuclear Safety Regulators Group
Stress Test Peer Review Board

1 GENERAL QUALITY OF NATIONAL REPORT AND NATIONAL ASSESSMENTS

The accident at the Fukushima nuclear power plant in Japan on 11th March 2011 triggered the need for a coordinated action at EU level to identify potential further improvements of Nuclear Power Plant (NPP) safety. On 25th March 2011, the European Council concluded that the safety of all EU nuclear plants should be reviewed, on the basis of comprehensive and transparent risk and safety assessments - the stress tests. The stress tests consist of three main steps: a self-assessment by licensees, followed by an independent review by the national regulatory bodies, and by a third phase of international peer reviews. The peer review also consists of 3 steps: an initial desktop review, three topical reviews in parallel (covering external initiating events, loss of electrical supply and loss of ultimate heat sink, and accident management). There are seventeen individual country peer reviews.

The Hungarian National Report (HNR) has been prepared on the basis of the Final Report of Paks NPP on Targeted Safety Re-assessment (TSR) of the plant. The re-assessment addressed the nuclear safety issues, with special regard to protection against external causes. All nuclear facilities (unit 1-4 including the spent fuel pools) located on Paks NPP site were covered. The report does not cover the facility for dry storage of spent fuel.

A draft country review report was prepared during the topical review based on the HNR and discussions on the three topics. It was delivered to Hungary in advance, to serve as a basis for the stress-test review team visit. The visit encompassed discussions with the regulatory body and the operator, and included a walk-down of the site. The basis for the discussion during the country visit was the list of open issues, upon which Hungary provided the information in advance of the mission. The topics for the site visit where delivered to Hungary in advance. During the plant visit, in addition to the clarification and explanations, the review team observed specific locations, equipment, arrangements and the use of procedures.

Hungary has submitted a comprehensive national report, specifying the analyses undertaken and results obtained. During the topical review Hungary provided extensive answers and explanations to questions that were raised. During the country visit both the regulator and the operator provided appropriate clarifications and justifications and access to required documentation. In the course of the plant visit all the locations requested by the peer review team were made accessible. Unhindered access has been granted and sufficient explanations provided. With this, the peer review team want to commend the regulator and the operator and in particular those colleagues who were directly involved in the review, for the good cooperation and mutual respect.

The Country Report was finalized at the end of the country review, and agreed between the peer review team and the national regulator. The main findings of the peer review are specified at the end of each topical chapter.

1.1 Compliance of the national reports with the topics defined in the ENSREG stress tests specifications

The HNR is compliant with the topics defined by ENSREG. The structure of the ENSREG specification is strictly followed, which facilitated the review and the understanding of the information. As a general comment, the report is well written and focuses on the key issues.

General conclusions are drawn and the needs for potential safety improvements or further analysis of Paks NPP are also covered.

1.2 Adequacy of the information supplied, consistency with the guidance provided by ENSREG

The information provided in the report, during the country review and country visit is adequate and is considered to be consistent with the guidance provided. The situation in the country was well presented and explained. Hungary provided a great deal of information, which was clearly and properly structured. Hungary has only one NPP site and the report is focused on it.

1.3 Adequacy of the assessment of compliance of the plants with their current licensing/safety case basis for the events within the scope of the stress tests

The report does not identify any cases where the plant is non-compliant with its license conditions in terms of the design basis earthquakes (DBE), floods (DBF) and extreme weather conditions, as well as in terms of protection from loss of power or heat sink.

At the time of construction the design basis, the plant did not include many requirements (such as ground motion level, return period for external events, etc.), that were added later on. Through the upgrading process the plant managed to comply with all of those.

In addition, at the time of construction there were no regulatory requirements for beyond design basis accidents. Now such requirements are in place and consequently the plant could comply with those following of adequate modifications.

As a pre-condition for the planned service life extension, the authority requires that the modifications necessary for the management of severe accidents shall be completed prior to the expiry of the original design lifetime for each unit.

1.4 Adequacy of the assessments of the robustness of the plants: situations taken into account to evaluate margins

The margins for seismic events and flooding are assessed. The evaluation of safety margins with respect to extreme weather conditions is not completed.

Comprehensive assessments of the robustness of the plant are made in all situations including Loss Of Off-Site Power (LOOP), Station Blackout (SBO) and Loss of of Ultimate Heat Sink (UHS). However, UHS + SBO was enveloped by SBO and therefore not specifically analyzed.

The HNR addresses all components which are considered essential for management of severe accidents and which are planned for implementation or are presently in the phase of discussion. Organizational arrangements for accident management and emergency planning, measures to install corresponding hardware for mitigation of severe accident (depressurization, hydrogen management, filtered venting or internal containment cooling, etc.), together with procedural arrangements (symptom based Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs)) have been addressed in the report.

1.5 Regulatory treatment applied to the actions and conclusions presented in national report

There is a clear evidence of the regulatory engagement in the stress tests performed by the plant. The regulatory body reviewed the analyses undertaken and concluded to be appropriate. In the process, regulatory inspections were conducted to check operator's compliance with ENSREG specifications for conducting stress test. Furthermore, inspections were undertaken to corroborate some of the findings in relation with safety margins and future improvements.

Upon completion of the self-assessment the operator proposed a series of measures to increase safety margins. The regulator has reviewed the proposed measures and concluded them to be relevant and appropriate for addressing the opportunities for improvement identified in the self-assessment. Nevertheless, the regulator determined that some additional measures are warranted.

The implementation of the measures to improve margins requires undertaking of complex analyses and implementation of various tasks. The regulator therefore required the preparation of an action plan that will contain the detailed specification of each task and the respective completion deadlines. This plan is to be submitted by June 30th, 2012. It will be subject to regulatory approval.

The regulator agrees with the licensee regarding the assessment of the safety of the plant after full performance of the proposed safety improvement measures:

- The probability of severe accidents affecting the reactor caused by long-term loss of electricity or of the UHS will decrease.
- By ensuring an alternative water supply and alternative electric supply options, severe accident situation at the spent fuel pool (SFP) might be prevented or mitigated.
- Extreme external events may still result in damage at the site but the consequence of the damages will be further reduced.
- Capability to prevent and/or mitigate accidents affecting several units at the site will increase.
- Emergency response solutions will be extended to cover multi-unit accidents.

2 PLANT(S) ASSESSMENT RELATIVE TO EARTHQUAKES, FLOODING AND OTHER EXTREME WEATHER CONDITIONS

2.1 Description of present situation of plants in country with respect to earthquake

2.1.1 DBE

2.1.1.1 Regulatory basis for safety assessment and regulatory oversight

The Hungarian Atomic Energy Authority (HAEA) is the legal authority supervising the safety of nuclear installations in Hungary according to the basic safety functions laid out in the Act CXVI of 1996 (plus amendments).

The nuclear safety requirements were detailed in the Govt. Decree 89/2005. (V. 5.) and in its Annex, in the Nuclear Safety Codes (NSC) and the latest update of the NSC by a governmental decree issued on August 10, 2011. The NSC prescribes, in general, the main frames and requirements that determine the design basis of the plant stating that hazards of natural origin with a recurrence frequency higher than 10^{-4}/year shall be accounted for in the design basis. The NSC requirements are in line with the international practice (IAEA safety standards and WENRA reference levels).

2.1.1.2 Derivation of DBE

The first deterministic Seismic Hazard Assessment (SHA) for the Paks site performed in the beginning of the 1970s' derived a design base intensity of I=6°MSK corresponding to Peak Ground Acceleration, PGA=0.025-0.05g by setting the level one intensity degree higher than the strongest observed earthquake.

Further evaluations of seismic hazard took place by using Probabilistic Seismic Hazard Assessment (PSHA) in accordance with IAEA safety standards and international practice. As a result, in 1996 the DBE PGAH=0.25g and PGAV=0.20g for the occurrence probability of 10^{-4}/year was established. Those values, corresponding to the SL-2 level, are still relevant today. Data corresponding to Seismic Level 1 (SL-1) are not mentioned in the report but have been explained during the discussions, as documented below.

2.1.1.3 Main requirements applied to this specific area

At the time of construction of the Hungarian NPP, Paks, in the 1970's, there were no requirements for the nuclear seismic design basis (buildings were designed to industrial-construction code). As a pre-condition for the planned service life extension, the authority required that the modifications necessary for the management of beyond design basis events and severe accidents shall be completed prior to the expiry of the original design lifetime of each given unit.

The current Hungarian regulation takes the 0.005 non-exceedance probability earthquake as design basis for the whole life span of the plant (which corresponds to 10^{-4}/year in case of a 50-year life time), the parameters of which (PGA, response spectrum) shall be determined based on the median hazard curve accounting for site effects.

The DBE for the plant was specified on the 10^{-4}/year exceedance level using the weighted mean hazard curve. From the report it appears that the Operating Basis Earthquake or SL-1, following IAEA standards, is still not defined. However during the country peer review it was clarified that according to the Hungarian regulatory requirements the level SL-1 for safe operation has to be set. In fact the SL-1 earthquake has been defined in accordance with the international and national requirements and international best practice and was documented in the Final Safety Analysis Report (FSAR).

2.1.1.4 Technical background for requirement, safety assessment and regulatory oversight

The PSHA leading to the current DBE assessment between 1986 and 1996 was performed in connection with a PHARE project of the European Commission supported by review missions of IAEA. The regulator states that SHA was done in accordance with international practice and IAEA standards and guidelines of that time. The most recent assessment uses an advanced PSHA with a logic-tree approach to model uncertainties related to the database and model assumptions.

In addition to probabilistic methods micro-seismic stress and movement measurements have been performed. Hazard analysis further includes a serious effort to assess potential capable faults in the site vicinity. Faults were analysed by seismic reflection profiles. In addition, the hazard of soil liquefaction has also been evaluated in depth.

2.1.1.5 Periodic safety reviews

In compliance with the Atomic Act, the licensee shall perform Periodic Safety Review (PSR) every 10 years, which took place in 2007-2008 the last time.
Seismic hazard assessment has been reviewed during PSRs in 1999 and 2007 updating the assessment in accordance with new scientific evidence and data obtained by a micro-seismic network monitoring of the site. The development of the state-of-the-art and international standards has been also taken into account during the PSRs. The PSR in 2007 has not resulted in modification of DBE. During the PSR the compliance of DBE definition and the hazard analysis with state-of-the-art requirements has been established. As a result of the latest PSR in 2007 some new findings were defined, namely: need for the modification of viscous medium and certain maintenance of the viscous dampers have been recognized; update of document archive database; improvement of classification database; initiation of some measures, resulted from the seismic PSA (liquefaction study) and other reviews (necessity for improvement of seismic housekeeping).

2.1.1.6 Conclusions on adequacy of design basis

The Hungarian regulator states that the seismic hazard of the Paks NPP site has been assessed according to the Hungarian requirements, in line with international standards and good practices. It is said that as a result of micro-seismic monitoring, the reliability of the input data of hazard calculations regarding the frequency of earthquakes has been significantly improved. The report provides sufficient information to enable the conclusion that the described methodology for deriving the current DBE for the Paks NPP is widely compliant with the current state of the art and the IAEA safety standards.

2.1.1.7 Compliance of plant(s) with current requirements for design basis

Due to the fact that seismic hazard has been underestimated during the sitting of the NPP Paks, the plant has not been designed to withstand earthquake loads. In order to qualify the plant for the upgraded DBE, a seismic safety-upgrading program has been started in 1993. The HNR describes in detail numerous reinforcement and upgrading measures including the reinforcement of buildings. A large number of important reinforcement and qualification measures were implemented during the time period from 1993 to 2003.

The report concludes that, by implementing reinforcement and upgrading measures the plant complies with the current seismic safety requirements. Nevertheless, some indirect effects of seismic events were identified, and additional measures implemented in order to improve the level of the seismic safety. Those included: upgrading or fixing of non-safety classified Structures, Systems and Components (SSCs), failure of which can jeopardise safety function; additional protection against seismic interactions (fire, internal flooding). In the frame of stress tests additional SSCs have been considered, e.g. building of the fire brigade.

2.1.2 Assessment of robustness of plants beyond the design basis

2.1.2.1 Approach used for safety margins assessment

Evaluation of safety margins was done by assessment of the impact of earthquakes of different magnitudes and spectra, up to those leading to severe fuel damage. The following approaches were used: (1) Simple empiric or semi-empiric methods. (2) Numerical modelling: safety margins with respect to soil liquefaction are tested using soil mechanics, empiric methodologies and stress calculations. (3) Probabilistic methods (seismic PSA: the safety margins for the loss of electrical power supply, the loss of UHS and the loss of containment function due to earthquakes are determined by probabilistic assessments constraining High Confidence of Low Probability of Failure (HCLPF) values). (4) Selected active safety components as well as control assemblies of the Reactor Protection System (RPS) were tested by performing shaking table experiments in order to identify their seismic margins.

2.1.2.2 Main results on safety margins and cliff edge effects

In HCLPF analyses the mean probability of occurrence reaches the value of 0.5 at certain acceleration, which is associated with an earthquake of a frequency of only 10^{-5}/year (which is lower than the plant's design basis of 10^{-4}/year). The range of calculated accelerations is the following: Electric power supply function: PGA=0.46g; UHS: PGA=0.42g; Loss of containment integrity: PGA=0.53g. The probabilities for the loss of electrical power supply, the loss of UHS and the loss of containment function are quantified as functions of the earthquake load expressed in terms of ground acceleration.

Quantitative assessments of the safety margins against soil liquefaction reveal only narrow margins (approximately 1.1 margin in the 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. The consequent necessity of re-qualifying underground lines and connections is stated in the report. There are measures identified compensating the potential adverse effects of the liquefaction, which can be developed just after completing the ongoing works.

2.1.2.3 Strong safety features and areas for safety improvement identified in the process

A large number of important re-evaluation, and consequential reinforcement and qualification measures were implemented during the time period from 1993 to 2003 in the frame of the seismic-safety upgrading program.

Review of the design basis seismic safety pointed out some indirect effects of the earthquake, by the elimination of which the level of safety can be further improved.

2.1.2.4 Possible measures to increase robustness

The Report lists numerous measures, which are envisaged to increase the plant's robustness against earthquakes including: (1) Measures to avoid failure originating from liquefaction effects and building settlement; (2) Measures that are directed towards the completion of seismic qualification of certain systems and components; (3) A review of the database containing the seismic safety classification of components.

The following priorities are identified: (1) Further strengthening of some selected reinforced concrete items. These items had no seismic qualification before and have no direct safety functions; (2) The necessity of an automatic reactor shutdown shall be reviewed and resolved as needed; (3) Measures to prevent failures of underground line structures and connections due to buildings settlement caused by liquefaction during an earthquake.

With respect to the issue of an automatic reactor shutdown it was further clarified that based on the stress tests results and the lessons learned from other countries the rational of installation of such a signal will be analysed and justified depending on the expected safety benefit.

2.1.2.5 Measures already decided or implemented by operators and/or required for follow-up by regulators

In addition to measures described in 2.1.2.4 the following measures are required by the Regulator: (1) Analyses of the influence of the lack of seismic qualification of the filter structures of the essential service water system on the UHS function. (2) Review of the database containing seismic safety classification in order to provide confidence that the classification is in agreement with the licensing documentation of seismic safety improvement modifications.

2.1.3 Peer review conclusions and recommendations specific to this area

The reviewers acknowledge the measures undertaken to upgrade the plant, which was originally not designed to withstand earthquakes, to its current standard. The combination of repeated SHAs and subsequent retrofitting measures are among the “best practices” identified during the Stress Tests.

The original DBE has been subsequently re-evaluated by using probabilistic methods.

Comparing the latest SHA assessments against current international standards and research results, as requested in the ENSREG specification, shows that the described methodology for deriving the DBE for the Paks NPP is compliant with the current state of the art.

The reviewers acknowledge the wealth of geological and seismological information obtained as supplements to the HNR. This data shows that the Regulator identified and is aware of the active Mid Hungarian Fault Zone as well as other Quaternary faults in the near-region of the plant. It was stated that activity of these faults is “very low”. This statement is based on the available quantitative data on the age, slip rate and seismic capacity of these faults, contributing to the site seismic hazard. Additional analyses of these faults are planned for the next PSR period.

The beyond design capability is described and discussed in the report and the safety margins are defined. Evaluation of safety margins was implemented through the use of deterministic and PSA methods. Since the seismic PSA was done for Paks NPP, fragilities for all failure modes, including liquefaction, are available. The ultimate quantification of seismic margins was expressed by the Core Damage Frequency (CDF) for seismic events.

Robustness of the plant against earthquakes has been significantly increased recently by implementation of seismic safety upgrading programme. In addition several safety upgrading measures are envisaged.

It is recommended to the Regulator to monitor the implementation of the measures for strengthening of the level of protection of the plant structures against liquefaction effects and soil settlement, as well as for the completion of seismic qualification of certain SSCs and a review of the database containing the seismic safety classification of components.

2.2 Description of present situation of plants in country with respect to flood

2.2.1 DBF

2.2.1.1 Regulatory basis for safety assessment and regulatory oversight

The nuclear safety requirements were detailed in the Govt. Decree 89/2005. (V. 5.) and in its Annex, in the NSCs and the latest update of the NSC by a governmental decree issued on August 10, 2011. The NSC prescribes, in general, the main frames and requirements that determine the design basis of the plant stating that hazards of natural origin with a recurrence frequency higher than 10^{-4}/year shall be accounted for in the design basis. The NSC requirements are in line with the international practice (IAEA safety standards and WENRA reference levels).

2.2.1.2 Derivation of DBF

In the report two reasons for site flooding have been considered: natural flow pattern of the Danube River and damage of the upstream structures (dams). DBF for the plant was specified on the 10^{-4}/year exceedence level. Assessment of different sources of flood (icy flood, dam break, ice-pack) in the course of the passing of the flood wave concluded that the highest calculated water level is Bf 96.14 m (above Baltic sea) in the vicinity of Paks NPP. The site elevation level (Bf 97.15 m) is above the highest calculated water level.

2.2.1.3 Main requirements applied to this specific area

According to the regulatory requirements the natural origin flooding conditions shall be determined with a recurrence frequency of once in 10,000 years. The report states that concerning hazards due to failure of man-made facilities (dams) even in the cases of 10^{-7}/year recurrence frequency shall be taken into account.

2.2.1.4 Technical background for requirement, safety assessment and regulatory oversight

The background of requirements for the safety assessment is based on deterministic and probabilistic approach. Evaluation of flooding hazards of natural origin was based on the statistical assessment of data collected at the local water-gauges (for the period 1916–1985). Detailed analysis has been performed for the purposes of the stress-tests using a hydrodynamic model.

2.2.1.5 Periodic safety reviews

The report mentioned that the last PSR (2007-2008) had determined that certain flooding (atmospheric) hazards were not systematically documented and analysed. With respect to flooding issues, considered during the PSRs, no particular information is presented in the report. Later it was clarified that the review of the flood hazard was part of the PSR. The design basis for flooding was reviewed and justified in the frame of the second PSR in 2007.

2.2.1.6 Conclusions on adequacy of design basis

The report claims that since the flood level less by 1 m the level of the site, the flooding hazard needs not be taken into consideration. However, the machine rooms of essential service water pumps in the water intake works are located below the DBF flood level. Flooding of the machine rooms can occur through wall penetrations, which are not provided with water sealing. (See corrective action in 2.2.2.4) The flooding hazard caused by extreme precipitation is discussed in Section 2.3 below.

2.2.1.7 Compliance of plant(s) with current requirements for design basis

The conclusion of the report is that regarding the protection against floods the plant complies with its design basis, which is consistent with the regulatory requirements. It is, however, stated that some actions have to be taken to strengthen the level of protection of the essential service water system in order to cope with newly defined threats of flooding.

2.2.2 Assessment of robustness of plants beyond the design basis

2.2.2.1 Approach used for safety margins assessment

Model calculations with conservative assumptions were performed for the estimation of safety margins against flooding.

2.2.2.2 Main results on safety margins and cliff edge effects

The report claims that since the platform level of the site is by 0.85 m higher than the formation level of the embankments around the plant, flooding of the site is not anticipated. Even the damage of the nearest water reservoir (Gabchikovo) could only cause a water level, which remains 1 m below the platform level. It was concluded that flooding cannot result in the loss of basic safety functions of the plant due to sufficient safety margin.

2.2.2.3 Strong safety features and areas for safety improvement identified in the process

A strong safety feature of the plant is its site ground elevation above maximum possible water level in case of flooding caused by high flow pattern of the Danube River or dam break.

Area of safety improvement identified in the process of the stress-tests is strengthening of the level of protection of the essential service water system.

2.2.2.4 Possible measures to increase robustness

Since the peak water level of the Danube River taken into account in the review is above of the level of the machine room of the essential service water pumps, it is reasonable to increase the availability of the pumps by the modification of penetrations of the machine room wall to water sealed design.

2.2.2.5 Measures already decided or implemented by operators and/or required for follow-up by regulators

The report claims that flooding of the plant can be practically excluded, so no specific measures already decided or implemented by operators are identified in that respect. Possible measures which would be required for follow-up by the Regulator are described in 2.2.2.4.

2.2.3 Peer review conclusions and recommendations specific to this area

With respect to DBF the approach used for the assessment appears to be reasonable and in compliance with the international standards.

For the beyond design basis flood some model calculations with conservative assumptions were performed for the derivation of safety margins. It was concluded that flooding cannot result in the loss of basic safety functions of the plant due to sufficient safety margin.

It is suggested to the Regulator to monitor the implementation of specific measures for strengthening of the level of protection of the essential service water system against flooding.

2.3 Description of present situation of plants in country with respect to extreme weather

2.3.1 DB Extreme Weather

2.3.1.1 Regulatory basis for safety assessment and regulatory oversight

The nuclear safety requirements were detailed in the Govt. Decree 89/2005. (V. 5.) and in its Annex, in the NSC and the latest update of the NSC by a governmental decree issued on August 10, 2011. The NSC prescribes, in general, the main frames and requirements that determine the design basis of the plant stating that hazards of natural origin with a recurrence frequency higher than 10^{-4}/year shall be accounted for in the design basis. The NSC requirements are in line with the international practice (IAEA safety standards and WENRA reference levels).

According to Section 4.117 of NSC, among extreme weather conditions "large wind blasts, precipitation, accumulated ice and snow barrages, lightning, extreme high and low temperatures and drought" shall be taken into account.

2.3.1.2 Derivation of extreme weather loads

A Gumbel approach of the extremities was used for defining measured extreme values and design values of the extreme weather conditions parameters of 10,000 year recurrence. Concerning low water-level and low runoff of the Danube statistical methods similar to those used for weather conditions were applied to determine low water-level of 10^{-4}/year recurrence frequency.

During the stress tests review a new hazard analysis was prepared on the basis of the latest meteorological data available. 10^{-7}/year recurrence frequencies with the various confidence levels were determined for each external hazard.

2.3.1.3 Main requirements applied to this specific area

The report identifies that the initiating events by natural phenomenon with less than 10^{-4}/year recurrence frequency can be screened out from the scope of postulated initiating events included in the design basis. Nevertheless, safety analysis shall take into account the external events with lower frequencies as well as probabilistic assessment shall be performed down to 10^{-7}/year frequency.

2.3.1.4 Technical background for requirement, safety assessment and regulatory oversight

The background for the requirements is based on deterministic and probabilistic approach to be applied in the derivation of extreme weather loads (Gumbel statistics). PSA methods were also implemented. As a result the events with 10^{-7}/year recurrence frequency and various confidence levels were determined for each external hazard.

2.3.1.5 Periodic safety reviews

The report provides information that the last PSR (2007-2008) already determined that meteorological hazards were not fully and systematically documented and analysed. The necessary corrective measures were determined and the implementation is in progress. In the frame and after the PSR in 2007, new studies have been launched for the definition of beyond design basis (BDB) values and hazard curves that have been used and/or extended in the frame of stress tests.

2.3.1.6 Conclusions on adequacy of design basis

The approach applied for the assessment of design bases appears to be similar to the international standards. Reassessment of the consequences of extreme weather conditions up to 10^{-7}/year recurrence frequencies is ongoing. However some extreme weather conditions (e.g. tornado) were not evaluated and the corresponding design bases were not defined. Later it was clarified that tornados are not accounted for in the design basis since they are screened out using the probability (annual frequency) screening criteria. Under specific conditions of Hungary the maximum straight wind loads are bounding the tornado loads for the probabilities of design base and well beyond.

2.3.1.7 Compliance of plant(s) with current requirements for design basis

Regarding the protection against extreme weather conditions the plant complies with its current design basis, which is consistent with the regulatory requirements.

2.3.2 Assessment of robustness of plants beyond the design basis

2.3.2.1 Approach used for safety margins assessment

In the frame of the measures determined during the last PSR in 2007-2008 a system engineering evaluation was conducted to determine the systems and building structures, in the design basis of which the effect of an external hazard should appear. The influence of the given external effect of the safety functions is systematically established. Subsequently an item-by-item verification of the documentation is carried out concerning the compliance with the design basis and completeness of documentation. This work is in progress and that is why the results are not presented in the report. An engineering evaluation has been completed only for the safety related electric power supply systems and the preliminary results are described in the report.

For environmental loads BDB, the vulnerability of the structures has been and will be assessed and evaluated via a comprehensive fragility analysis and expert judgment.

2.3.2.2 Main results on safety margins and cliff edge effects

As described in 2.3.2.1 above no margin results are currently available with one exception for the systems of safety related electric power supply (e.g. the emergency diesel generators (EDGs)), which are the following: (1) Extreme wind load: no direct effect on the EDG buildings, however the dust protection on certain electric devices and safety related cabinets has to be further examined. (2) Extremely high temperature: maximum service temperatures have been defined for all the premises and devices. Mobile air-conditioners were ordered for certain sensitive equipment with intrinsic heat generation when existing margins are not adequate. (3) Extremely low temperature: lower limits of the service temperature have been defined. Some conditions are defined when temporary heating would be required. (4) Extreme rain, precipitation: under review, no margin analysis currently available. Extreme frost deposition and freezing rain are to be considered in the probabilistic analyses. (5) Lightning: protection provided according to the design standards. Overvoltage protection against the electromagnetic impulse of lightning provided only for recently installed equipment. (6) Drought: Recent examination of low probability events shows that the design bases for the lowest level of the Danube River is not violated.

Comprehensive hydraulic modelling of the drainage system is going on. Improvements of the drainage system are expected for beyond design basis precipitation and snow melt.

The main conclusion is that long timescales and availability of weather forecasts allow for remedial actions.

It was determined by studying the vulnerability curves that the margins are significant. No cliff edge effects are considered in the report due to the assumption that a small variation of any of the meteorological parameters would cause sudden failure. However, in the report is not provided specific information about the safety margins between the threats and design values for the different extreme weather conditions.

2.3.2.3 Strong safety features and areas for safety improvement identified in the process

Strong safety features of the plant with respect to the extreme weather conditions have not been presented in the report. The following areas of improvements are identified: qualification of buildings, completion of PSA assessments, and completion of engineering evaluation of safety margins.

2.3.2.4 Possible measures to increase robustness

The following measures are mentioned in the report: (1) Extreme precipitation: modification of the wall penetrations of the machine building of the essential service water system pumps to a sealed design in order to increase robustness against flooding; (2) Extreme weather conditions: completion of the buildings qualification and implementation of the necessary reinforcements.

2.3.2.5 *Measures already decided or implemented by operators and/or required for follow-up by regulators*

In addition to measures listed above the Regulatory body considered necessary to require the following measure with regard to lightning protection: Classification of system components important to safety which are endangered by electromagnetic effects (including lightning induced).

2.3.3 Peer review conclusions and recommendations specific to this area

With respect to extreme weather loads, the approach used for the assessment appears to be reasonable and in compliance with the international standards. In the derivation of extreme weather loads both deterministic approach and PSA methods were used.

The BDB capability is described and discussed in the report and the safety margins are defined. The vulnerability of structures with respect to beyond design basis loads has been assessed and evaluated via comprehensive fragility analysis and expert judgment. However, the report does not contain specific information about the numerical values of the safety margins of the extreme weather conditions parameters. Special attention should be paid for defining vulnerability of the rain drainage system in case of BDB of extreme precipitation and snowmelt. The HNR indicates that the work is going on with respect to the vulnerability assessment of extreme weather conditions including the rain drainage system.

The report provides satisfactory evidence for a PSR process being applied for the assessment of certain meteorological hazards.

It is suggested to the Regulator to monitor the implementation of specific measures for strengthening of the level of protection of the plant SSCs against extreme weather conditions.

3 PLANT(S) ASSESSMENT RELATIVE TO LOSS OF ELECTRICAL POWER AND LOSS OF ULTIMATE HEAT SINK

3.1 Description of present situation of plants in country

3.1.1 Regulatory basis for safety assessment and regulatory oversight

The legal framework of the use of atomic energy is laid down in the Act CXVI of 1996 on Atomic Energy, which has been amended many times since issuing. The Act, among others, determines the basic safety functions.

The nuclear safety requirements were detailed in the Govt. Decree 89/2005. (V. 5.) and in its Annex, in the NSC and in the latest update of the NSC by a governmental decree issued on August 10, 2011. The new regulation did not essentially change the requirements that are applicable to this stress test topic.

3.1.2 Main requirement applied to this specific area

As defined in the NSC, redundancy, diversity, single failure criteria, etc. are required for all safety systems. The most general design requirements related to the heat removal and power supply include:

- Heat generation and heat transfer phenomena shall be determined and analyzed. Heat removal to the UHS shall be ensured. In addition, passive means for heat removal should also be considered.
- It shall be ensured that the electric power supply system is able to supply the electric power to operate the safety related systems and components during normal operation, anticipated operational occurrences and design basis accidents.
- Safety related electric power supply systems and components shall be designed according their safety class.

3.1.3 Technical background for requirement, safety assessment and regulatory oversight

Nuclear safety licensing in Hungary is based on deterministic analyses, but the regulations require the implementation of probabilistic analyses too. In line with that, Paks NPP has completed Level 1 and Level 2 PSA studies in relation to the accidents on both reactor and SFP, for all power and shutdown modes, including internal initiators and hazards, as well as seismic PSA.

The safety requirements were compiled on the basis of the IAEA safety standards. One of the main goals of the recent regulatory requirement review has been to ensure the full correspondence to the WENRA reference levels. In addition to the most recent IAEA requirement documents, the WENRA reference levels provided the main sources for the reviewers. Due to this process it can be stated that current requirements are expected to be fully compliant with the WENRA reference levels shortly.

3.1.4 Periodic safety reviews

Within the PSR in 2007, several safety improvements actions were identified, which were under implementation phase at the time of the Fukushima accident.

3.1.5 Compliance of plants with current requirements

The review process found no indication that Paks NPP is not in compliance with the current requirements.

It is noted that HAEA has launched a general inspection procedure including specific inspections in the process of review of the targeted reassessment activity of the licensee to verify the compliance.

3.2 Assessment of robustness of plants

3.2.1 Approach used for safety margins assessment

The national report meets the ENSREG Specifications. It describes in detail the design provisions of the plant electrical systems for Paks NPP design and shows capabilities to cope with loss of off-site power, SBO and loss of UHS.

The general approach adopted in assessing the safety margins with respect to the loss of electrical power and loss of heat sink aspects of the stress tests requirements is to identify associated level of redundancy and diversity as well as the timescales by which various safety functions need to be implemented in order to prevent significant fuel damage.

3.2.2 Main results on safety margins and cliff edge effects

3.2.2.1 Loss of Off-site Power (LOOP)

A design solution for LOOP is to transfer the plant to the house load operation; if it is not possible, the plant safety buses are powered from the EDGs. Each train of the emergency power supply system is capable of ensuring safe shutdown state in all design basis accidents. In case of LOOP, signals for start-up of the diesel generators are actuated independently for each EDG. Each of the 12 EDG (3 per unit) is equipped with 100 m3 underground fuel tank that contains 70 m3 of fuel sufficient for 120 hours operation. The fuel is stored in tanks protected against earthquake and flooding. The operation period of the EDGs without interventions can be extended by almost 30% through the increase of the stored quantity of fuel. The EDGs are cooled by the essential service water system. Without this system the EDGs can not operate; however the cooling water can also be provided from the fire water system.

In the case of loss of normal power supply, the batteries are capable to supply electrical power to the measurements and for the implementation of the required intervention for 3.5 hours. The EDGs starting meantime will charge the batteries.

As an alternative power supply options in case of availability of the relevant power transmission lines and switch stations, Paks NPP can be supplied from Dunamenti Gas Turbine Plant through the 120 kV grid, or from the Liter Gas Turbine Plant through the 400 kV grid. Based on accomplished tests the arrangement of the dedicated supply route and the required switching operations can be performed within one hour. Dunamenti Gas Turbine Plant has autonomous power source, which makes possible its start-up without external power supply or the national grid.

3.2.2.2 Station Blackout (SBO)

In case of station blackout only those battery supplied and compressed air operated systems are available, which provide energy to the measurements and the most important interventions. The secondary side pressure can be stabilized and reduced by the proper application of the emergency operating instruction and opening of the pressure reduction valves to the atmosphere. Water can be supplied to the steam generators at lower pressure through an alternative supply route. Detailed additional information on manual option to open steam dump to atmosphere valves and operability of air-operated valves was provided during the country visit.

In the case of SBO occurring during operation at normal power, without electrical power supply and secondary side alternative supply, the steam generators dry-up within 4.5 hours after the loss of power, the heat removal gets lost and core damage may occur in about 10 hours after the loss of power.

Without electrical power supply the circulation of the cooling water stops in the spent fuel pool. Assuming the most conservative configuration with the highest residual heat and normal water level, which is much lower than the refuelling water level, an intensive boiling may start after 4 hours as soonest. Damage to the cladding of the fuel assemblies may commence after about 19 hours (this period might be 25 hours, if the water level is higher).

In the case of SBO, mobile severe accident diesel generators are available, capable to supply measuring, controlling and intervention systems that can be used for the implementation of severe accident prevention and mitigation measures (including supply of accumulator batteries). Due to their limited capability to supply electrical power to safety supply systems and to the essential service water pumps, among the corrective measures is decided to supply additional, diverse diesel generators to manage accident situations.

Detailed additional information on experimental tests made by the pump manufacturer that confirmed tightness of main coolant pumps seals was provided during the country visit. It was confirmed that Main Coolant Pump (MCP) seals leak is not a concern for Paks NPP.

3.2.2.3 Ultimate Heat Sink (UHS)

The UHS at WWER-440 units is provided via the essential service water system (ESWS). If it is unavailable, the secondary feed & bleed via Steam Generator (SG) may be initiated; however during shutdown states, the secondary feed & bleed is less effective and requires lot of feedwater reserves. Design features related to prevention of loss of essential service water system include a three-train design, each with 100% cooling capacity calculated for DBA, independence of the system from external supplies and physical separation.

During accidents with loss of heat sink, the emergency water to the secondary circuit can be supplied by the demineralised water system, through seismic reinforced system components. Depending on the accident conditions, the cooling may be performed using the emergency feedwater pumps or the auxiliary emergency feedwater pumps, which supply water directly from the main pipeline of the demineralised water tanks to the steam generators. For shutdown state, the demineralised water tank inventory might be sufficient to provide essential service water for 2 to 3 days.

Depending on the power supply availability, the alternate cooling options include the following paths:

- Bank filtered wells – technology cooling water system – essential service water system;

- Plant fire water pump station – technology cooling water system – essential service water system;
- Diesel fire water pumps – technology (non-essential) cooling water system – essential service water system;
- Fire fighter vehicles, Diesel pumps in cascade;
- Demineralised water tank park;
- Essential service water system /fire water system – technology (non-essential) cooling water system – make-up water softener – demineralised water tank park

The probability to lose the ultimate heat sink of the spent fuel pool is low. It can be concluded that the fuel damage will commence after 10 hours (as a minimum, depending on the situation) following an accident occurring during power operation; the fuel damage starts after 19 hours (as minimum) in SFP.

3.2.2.4 Loss of UHS & SBO

The case of a combined SBO and loss of UHS in case of Paks NPP is in fact covered by the station blackout, since the station blackout is always connected with the loss of UHS.

3.2.3 Strong safety features and areas for safety improvement identified in the process

The inherent safety features of WWER-440/V213 plants contributing to significant time margins in case of loss of electric power and loss of UHS include large thermal inertia due to low power and comparably large amount of water both in primary and secondary system, as well as large volume of water inside the containment stored in the pressure suppression system potentially available for cooling of the core.

In the case of disturbance in or LOOP grid, as the first level of defence in depth the units are automatically separated from the national grid to island service mode and controlled to in-house load level. This reduced power level of only one unit is still sufficient to supply the needed electric power to the in-house consumers of all four units.

The EDGs compose three totally independent trains having identical construction. In addition to the cooling with essential service water, the DGs could be cooled by the fire hydrant and the connecting point is easy accessible to get cooling quickly.

In case of loss of ultimate heat sink a lot of alternative cooling options are possible. These assure low probability to lose cooling capabilities. One of the interesting solutions is possibility of using discharge water canal for intake of fire water pumps in case of unavailability of emergency water system.

3.2.4 Possible measures to increase robustness

As described above, the evaluation of safety margins at LOOP and SBO proved the ability to ensure protection of safety barriers during considerable period of time, thus providing sufficient time for accident management actions for recovery of the plant power supply. Despite this, the robustness of the current plant design will be enhanced by planned measures, specified in Section 3.2.5 below.

3.2.5 Measures already decided or implemented by operators and/or required for follow-up by regulators

The following measures have been envisaged to increase the robustness of the plants in case of loss of electrical power:

1. The protection of 400 kV and 120 kV substations and of the automatic switch to island service mode will be evaluated against earthquakes, and improved as appropriate.
2. In addition to the existing severe accident diesel generators supplying electrical power to measurement and control systems described in accident management procedures, diverse diesel generator, which can supply electrical power to safety consumers having role in severe accident prevention and long term accident management is being considered. 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 required consumers, pumps and valves. The number and capacity of the diverse accident diesel generators have to be determined.

3. With regard to on-site Alternating Current (AC) power sources, operating instructions have to be developed to manage potential, alternative and not yet used supply routes between the normal reserve and the safety trains of the units as identified during the re-assessment.
4. 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.
5. The black start ability of the gas turbine located in Litér will be assured by installing a diesel generator.
For enhanced resistance of the plant in the case of loss of UHS the following modifications are planned:
6. The safety electrical power supply of band screens have to be solved in order to prevent the blockage of the screens of the essential service water system.
7. Comprehensive inspection, maintenance and operational testing shall be introduced regarding those equipment that are to be applied in the frame of actions planned to be implemented in case of low water level. The still missing inspection, testing and maintenance procedures have to be prepared.
8. In order to guarantee the availability of the potentially jeopardised three demineralised water tanks of Installation II (units 3 and 4), the covering panels of the service building have to be qualified or the tanks have to be protected against the DBE.
9. The operator has to maximize the continuously available inventory of the stored demineralised water by the modification of the Technical Specifications and the operating instructions, with the consideration of the free storage volume of the demineralised water tanks.
10. The water base of the earthquake resistant fire water pump station of Installation II that is equipped with individual diesel-driven pumps and capable to operate for eight hours can be utilized only if the essential service water systems are operating. The accessibility of the 2x2000 m3 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.
11. The electrical power supply of the submersible pumps of the bank filtered well plant has to be established by a well protected fix or mobile diesel generator in order to guarantee their applicability in severe accident situations.
12. Similar to the connection existing on Installation I (units 1 and 2), the issue of water supply has to be solved from the fire water system to the essential service water system through the technology cooling water system.
13. The equipment necessary for the cooling water supply to at least one EDG of each unit from the fire water system have to be available; so as the EDG can be started and operated in case of loss of the essential service water. Operating instruction has to be completed with the measures to be implemented for the application of this alternative cooling.
14. Connection points have to be established on the demineralised water tanks to allow the water supply, through the auxiliary emergency feedwater system, by mobile equipment.
15. Based on the existing potential direct cooling water supply to the containment, the issue regarding supply of water containing sufficient boron acid has to be solved with the use of the existing tanks. The potential setting of the boron concentration of water inventories from external sources has to be solved. The supply mode from external source to the containment has to be regulated in an operating instruction.
16. 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 potential 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.
17. The access to the connection point of the auxiliary emergency feedwater system established for external water supply and to the valves required for its operation under accident conditions has to be reviewed, and modified if needed.
Proposed additional measure judged necessary by the authority.
18. A PSA has to be performed to assess whether an operating time limit for a closed reactor being below 150 ^{o}C temperature is reasonable to be established and introduced.

3.3 Peer review conclusions and recommendations specific to this area

The information presented in the national report is comprehensive and covers the areas of requirements under ENSREG specifications. The plant description is supported with simplified diagrams, which facilitate better understanding of the system features. The electrical supply systems and equipment ensuring UHS are analysed adequately and the results are confident. Deficiencies identified are covered by proposed improvement measures.

Many measures are already implemented. Paks NPP has some capabilities for cooling following the loss of UHS. One of the interesting solutions is the possibility of using discharge water canal for water intake of fire water pumps, which could in turn supply essential service water system.

The possibilities of interconnection of existing equipment are beneficial. However might also lead to loss of separation. Such improvements or modifications should be prepared carefully. Before the implementation, separation issues should be investigated.

The switchyard is partially seismically reinforced and the plant also tries to reinforce other non-safety equipment.

4 PLANT(S) ASSESSMENT RELATIVE TO SEVERE ACCIDENT MANAGEMENT

4.1 Description of present situation of plants in Country

4.1.1 Regulatory basis for safety assessment and regulatory oversight

The nuclear safety requirements were detailed in the Govt. Decree 89/2005. (V. 5.) and in its Annex, in the NSC and the latest update of the NSC by a governmental decree issued on August 10, 2011. The NSC prescribes, in general, the main frames and requirements that determine the design basis of the plant. These requirements are in compliance with international practice (IAEA Safety Standards and WENRA Reference Levels).

4.1.2 Main requirements applied to this specific area

The implementation of SAM is required by the regulatory authority. Regulatory requirements on SAM have been in place since before the Fukushima accident. HAEA has adopted the WENRA Reference Levels, which include requirements on design basis extension, SAM and emergency preparedness. On Unit 1, the SAMG were developed, trained and introduced by the end of 2011. The hardware modifications for SAM were also implemented for Unit 1. For all other units of the Paks plant, the implementation of severe accident measures and SAMG are scheduled up to the end of 2014.

4.1.3 Technical background for requirement, safety assessment and regulatory oversight

Nuclear safety licensing in Hungary as specified in the NSC is based on deterministic analyses. Additional regulations require the implementation of probabilistic analyses as well. In line with that, Paks NPP has completed Level 1 and Level 2 PSA studies including the accidents of the reactor and of the SFP. PSA analyses cover the normal operating mode and the low power and shutdown mode. The Paks NPP strategy of SAM was determined on the basis of Level 2 PSA results. However, hardware modifications addressing the management of severe accidents have not yet been incorporated in the Level 2 PSA.

4.1.4 Periodic safety reviews

During the PSR performed in 2007, several safety improving measures were decided. Their implementation was partially completed or in progress at the time of the Fukushima accident.

Following the self-assessment in the stress tests, and in agreement with the regulator, the implementation of these measures was accelerated.

4.1.5 Compliance of plants with current requirements

It was stated by the regulator that WENRA Reference Levels relevant to accident management are now established in Hungarian regulations. The EOPs are implemented and consist of symptom oriented operating procedures. Also the SAMGs seem to follow the current international practice. For Unit 1 of Paks NPP, the implementation of the SAMGs has been finalized in 2011. Hardware modifications for SAMGs are currently being implemented step by step for Units 2, 3 and 4 with a completion deadline of 2014.

4.2 Assessment of robustness of plants

4.2.1 Adequacy of present organizations, operational and design provisions

4.2.1.1 Organization and arrangements of the licensee to manage accidents

After the declaration state of emergency, the Emergency Response Organization (ERO) of the Paks NPP is operated in line with the national regulations and plant level internal procedures. The documents regulating the activity of the ERO contain the detailed rules of use of the mobile equipment and reserves available. These documents include the preparatory and inspections tasks, as well as the practical actions in connection with the elimination of the emergency situation. The scope of the ERO includes the design basis accidents and severe accidents.

The NSC requires that the licensee have appropriate (objective organizational and administrative) preparedness to mitigate the consequences of BDB events. The report provides only brief information on the organization and arrangements of the licensee to manage accidents. Further details have been provided in the presentation delivered during the topical and country review.

The operating staff performs its activity according to operating procedures. In the case of a severe accident, subsequent to the activation of the ERO, the structure of operative control remains unchanged until the SAMGs have to be applied. During the implementation of SAMGs the operative control over the personnel of the affected unit is taken by the ERO in the Technical Support Centre (TSC) from the plant shift supervisor. In such a situation the personnel of the damaged unit perform the activity based on instruction, which are directly announced to the control room by the TSC working in the Protected Command Centre (PCC).

The control facilities (Control Rooms (CRs), PCC) seem to be well prepared for the SAM conditions. If the CR is inhabitable, then the reserve CR is at the disposal of the staff. In the reserve CR the actions needed for shutting down and cooling down the unit, as well as those needed for the maintenance of the cold state can be executed. If none of the CRs are habitable, then the control room personnel keep contact with the TSC from predefined sheltered locations outside the main building.

With the existing organizational structure and facilities it seems to be adequate to cope with a severe accident on one single unit. Arrangements to provide tools and resources for the response to a multi-unit accident do not currently exist at the plant. In case of a multi-unit accident, the performance of the emergency response tasks would require the help of the national competent organizations. The analysis of the resources necessary for multi-unit events has started but was not finalized at the time of the review. The plans for improving emergency response should take account of the need for training from the necessary external resources to be involved in fire fighting and technical and medical rescue. Currently these are not trained for response to a multi-unit event. It could not be assessed to which extend those external resources are helpful in severe situations, especially in case of destroyed infrastructure. There are different agreements for supporting the NPP staff in case of a severe accident.

4.2.1.2 Procedures and guidelines for accident management

EOPs and SAMGs have been developed for all operating modes (normal operating and shutdown), for SFP accidents in all units. The SAM documentation was completed by the end of 2011. In Unit 1 the SAMG package has been introduced and all related hardware modifications have also been implemented already. For the other three units SAMGs will formally be implemented after all hardware modifications are completed.

4.2.1.3 Hardware provisions for severe accident management

The hardware modifications for the NPP are following the principal elements of SAM:

external cooling of the reactor pressure vessel by discharging water from the bubble condenser and flooding the reactor cavity,

- severe accident management monitoring system,
- severe accident diesel generators for supplying electrical power to SAM designated consumers,
- hydrogen management under severe accident conditions by passive autocatalytic recombiners,
- prevention of coolant loss from the SFP due to pipeline rupture.

The state of implementation and the program for further planed implementation is listed below:

Measure	Unit 1	Unit 2	Unit 3	Unit 4
Plant changes for flooding the reactor vessel cavity	Implemented	2012 main outage	2013 main outage	2014 main outage
Provision of an autonomous power supply to designated consumers	Implemented	Implemented	Implemented	Implemented
Installation of passive hydrogen recombiners	Implemented	Implemented	Implemented	Implemented
Reinforcement of the spent fuel pool cooling system against loss of coolant	Implemented	Nov-Dec 2012	Feb-Mar 2013	Implemented
Installation of a severe accident monitoring system	Implemented	Jun-Aug 2012	Sep-Oct 2013	May-Jun 2013
Introduction of severe accident management guidelines	Implemented	31 Dec 2012	31 Dec 2013	31 Dec 2014

The modifications are pre-condition for the life-time extension of the plant as required by the regulatory authority.

Independently of the safety electrical power supply system, one mobile severe accident diesel generator is available for each unit. These generators are installed on platform trailers. They are stored in an earthquake protected building on site. These diesel generators can supply electrical power in the case of SBO to the monitoring systems which are needed for preventive actions to mitigate the consequences of the severe accident (e.g. pressure reduction of the primary circuit, flooding of the reactor cavity, SG relief within the hermetic compartments). The batteries can supply electrical power to the measurement system for 3.5 hours. This time should be sufficient to put the severe accident diesel generators into position and start the operation.

The plant fire fighters are able to perform the water intake from the Danube canal and to supply water to the connection of the auxiliary emergency feedwater system in sufficient quantity and on adequate pressure by the tools at their disposal. This alternative cooling water source can be established directly from the intake water canal of the Danube or from the fish lakes containing about one million cubic meters of water.

4.2.1.4 Accident management for events in the spent fuel pools

The SFP of each unit is located in the reactor hall, outside the containment. The fuel assemblies might be stored on two layers. The SFPs have no second independent water supply or an additional external water supply. The quantity and possible release of hydrogen in the reactor hall during an accident occurring simultaneously on two damaged SFPls and two (one open and one closed) damaged reactors in a twin unit has to be analyzed. According to the valid emergency operating instruction, the water make-up (without electrical source) could be provided to the SFP by the gravity-forced discharge of water from the upper trays of the bubble condenser, while the lower trays being used for reactor cavity flooding. There is no confirmation so far that the amount of water in the bubble condenser would be sufficient for events affecting at the same time the reactor and the SFP. In order to improve safety, in the case of permanent loss of the UHS, the licensee plans to implement a corrective measure assuring the long term cooling of the SFPs by the establishment of a new, independent and protected supply route. For the SFP a PSA level 2 was performed. By ensuring an alternative water supply and alternative electric supply, severe accident situation of the spent fuel storage pool might be managed.

4.2.1.5 Evaluation of factors that may impede accident management and capability to severe accident management in multiple units case

For Paks NPP multi-unit accidents were investigated and it was pointed out that currently there are no adequate resources available to respond to a multi-unit accident. Such an event cannot be managed by the resources available on the site. The necessary external human resources to be involved are not trained for severe accidents occurring on several units at the same time.

The assessment of the radiological conditions in case of severe accident needs to be performed to address the situation of multi-unit events. The assessment has already been required by the regulator. In severe accident conditions, the access routes to control rooms and other locations where intervention is needed are determined by the ERO on the basis of conditions occurring during an accident. This information is provided to the personnel through the available communication channels.

The Backup Command Centre has also to be protected against earthquakes, radiation, external temperature, etc., and has to be equipped with the same controlling and communication systems like the Protected Command Centre. The technical support centre in the protected command centre has also to be extended to manage multi-unit accidents.

All described additional measures are still under consideration and will have to be implemented. The regulatory body obliged operator to provide a detailed action plan with tentative time-frames for implementation of the various actions. It has to be submitted to the regulator by the 30th of June 2012.

4.2.2 Margins, cliff edge effects and areas for improvements

4.2.2.1 Strong points, good practices

The agreement between the utility and the regulatory authority to update the PSA annually is commendable.

Regulatory requirement for SAMG implementation and for the installation of hardware for severe accident as pre-condition for life time extension is a strong point.

The arrangements in place in the PCC are commendable.

The requirement of SAMG in the national regulatory framework is commendable.

It is recognised that the implementation of measures for SAM started long time before Fukushima accident.

4.2.2.2 *Weak points, deficiencies (areas for improvements)*

In general, the Stress Tests Review did not identify major weak points for this topic. The Hungarian approach to manage severe accidents seems to be comprehensive. Nevertheless, there are areas where further improvement may be achieved:

– Full coverage of the issues associated with multi unit accidents including severe damage to the infrastructure, and the issue of generation and distribution of hydrogen in the reactor hall during twin-unit accidents.

– Upgrading the BCC against earthquakes, radiation, external temperature, etc., and ensuring that the BCC is equipped with the same controlling and communication capabilities as those at the PCC.

– An analysis of the long-term (beyond 1 week) severe accident consequences was carried out. Based on the analysis results suitable measures to prevent over-pressurization of the containment have to be developed and implemented. This should be realized with filtered venting or additional measures for internal containment cooling. These measures for long term internal containment cooling are considered to be adequate only in case of successful in vessel retention of the molten core.

4.2.3 Possible measures to increase robustness

4.2.3.1 *Upgrading of the plants since the original design*

It is known that at Paks NPP a certain number of upgrading measures were implemented in order to resolve safety issues, which were formulated in 1996 in frame of the IAEA extra budgetary programme on the Safety of WWER and RBMK NPPs. Nearly all issues were addressed and closed.

Some severe accident modifications have been completed, while others are still in the process of implementation. Also the outcomes of the annually updated PSA were taken into account in the SAM programme. The completion of the programme is a pre-condition required by HAEA for lifetime extension.

Due to the original design and limited preventive measures for BDB events, the main approach to mitigate severe accidents and their consequences at Paks NPP is to perform actions according to SAMGs. These SAMGs have been developed using results of investigations (including PSA Levels 1 and 2) started many years ago.

4.2.3.2 *Ongoing upgrading programmes in the area of accident management*

Ongoing upgrading (improvement) measures were started with the IAEA extra-budgetary programme on the Safety of WWER and RBMK types and complemented with severe accidents mitigation measures, which were already developed before the Fukushima accidents. A thorough analysis and modification project was launched in 2008 to mitigate the consequences of accidents BDB, which lead to severe damage to the reactor core. As a result of the project, several hardware modifications were needed for the introduction of the SAM. However the completion is in progress and performed unit by unit and should be finalized in 2014.

4.2.4 New initiatives from operators and others, and requirements or follow up actions (including further studies) from Regulatory Authorities: modifications, further studies, decisions regarding operation of plants

4.2.4.1 *Upgrading programmes initiated/accelerated after Fukushima*

The programme on development and implementation of hardware measures for severe accident mitigation measures and of SAMGs was started before the Fukushima accident and is still ongoing. As part of this programme, the installation process of hydrogen recombiners was accelerated after Fukushima accident.

The SAM documentation was completed by the end of 2011 in the frame of the SAM project; the actions are completed on Unit 1; the first trainings on the use of SAM guidelines were conducted in 2011.
The most important measures planned for implementation in regard to strengthening the site organization for accident management are as follows:

– The ways to guarantee conditions for radio communication have to be assessed in the case of permanent loss of electric power and earthquakes.
– Informatics mirror storage computers have to be installed both at the PCC and the BCC containing the necessary data (i.e. documentation, personal data, etc).
– The procedures for gathering the ERO personnel and for their transportation to the site have to be developed; the required equipment has to be identified and their provisions have to be established. A transportation vehicle providing adequate radiation protection under severe radiation conditions has to be purchased.
– The physical arrangement and instrumentation of the TSC established at the PCC have to be extended to provide sufficient resources for simultaneous management of severe accidents occurring on more than one (even all) units.
– The structure of the organization responding to accidents affecting multi units and the number of staff have to be determined; procedures have to be developed for personnel and equipment provisions, as well as for shift changes.

For all these planned measures, it has to be taken into account that the off-site infrastructure could be destroyed.

4.2.4.2 Further studies envisaged

Further studies on SAM are planned:

– Hydrogen generation and distribution in the reactor hall
– Long-progression with containment pressurization during severe accidents
– Updating the Level 2 PSA studies
– Development of a software based severe accident simulator.

The Paks NPP strategy of SAM was determined on the basis of Level 2 PSA results. However, current hardware modifications addressing the management of severe accidents have not yet been incorporated in the Level 2 PSA.

HAEA requested to develop more detailed studies on the following topics:

– The water supply to the SFP from an external source has to be made possible by additional connection from outside. This pipeline should be designed against external hazards. Water with boron concentration has to be supplied through this line to the SFP. The operating instructions have to be developed.
– Liquid radioactive waste management procedures have to be developed for severe accident situations. The SAMG have to be developed also to manage simultaneous accidents in the reactor and in the SFP.
– Analyses have to be carried out in order to determine the quantity and distribution of hydrogen in the reactor hall during an accident that simultaneously assumes in two damaged SFPs and two (one open and one closed) damaged reactors within a twin unit.
– An analysis of the long-term (beyond 1 week) severe accident consequences was carried out. Based on the analysis results suitable measures to prevent over-pressurisation of the containment have to be developed and implemented. This should be realized with filtered venting or additional measures for internal containment cooling. These measures for long-term internal containment cooling are considered to be adequate only in case of successful in vessel retention of the molten core.
– The analysis of radiological conditions in case of severe accidents, particularly for the case of multi-unit accidents has to be carried out.
– The resources needed for the on-site management of multi-unit accidents have to be analyzed.

4.2.4.3 Decisions regarding future operation of plants

As a pre-condition for the lifetime extension the regulatory authority required that the modifications for managing BDB events and severe accidents are necessary to be completed. This SAM programme is in progress for years and was completed in 2011 on unit 1, and will be completed in the other units according to the schedule by 2014.

4.3 Peer review conclusions and recommendations specific to this area

The decision of the regulatory authority to require implementation of SAM measures as pre-conditions for the life extension for all units is commendable. It is also recognized that Paks NPP has already implemented several accident management provisions as well as having developed the SAMGs. Also, the requirement by the regulatory authority to update the PSA annually is commendable.

The on-site organization and management of events, especially of multi-unit accidents, has to be improved. The Backup Command Centre has to be protected against earthquakes, radiation, external temperature, etc., and equipped with the same controlling and communication capabilities as those at the Protected Command Centre.

To avoid the release of radioactive material to the environment in case of long term of severe accident and to avoid over-pressurization a filtered containment venting system or a specific containment cooling system should be installed at all units. These measures for long term internal containment cooling are considered to be adequate only in the case of a successful in vessel retention of the molten core.

For further studies the regulatory authority mentioned the following points:

- Hydrogen generation and distribution in the reactor hall
- Updating the PSA level 2 and incorporating SAMG hardware modification
- Development of a software based severe accident simulator.

More detailed studies on the following topics are supported by the conclusion of the peer review:

- Water supply to the SFP from an external source has to be made possible by pipeline having adequate design against external hazards, with additional connection from outside. Water with boron concentration has to be supplied through this line to the SFP. The operating instructions have to be developed.
- Liquid radioactive waste management procedures have to be developed for severe accident situations.
- The SAMGs have to be developed also to manage simultaneous accidents in the reactor and SFP.
- Analyses have to be carried out in order to determine the quantity and distribution of hydrogen in the reactor hall during an accident that simultaneously assumes in two damaged SFPs and two (one open and one closed) damaged reactors within a twin unit.
- An analysis of the long-term (beyond 1 week) severe accident consequences was carried out. Based on the analysis results suitable measures to prevent over-pressurisation of the containment and release of radioactive material have to be developed and implemented. This should be realized with filtered venting or additional measures for internal containment cooling. These measures for long term internal containment cooling are considered to be adequate only in the case of successful in vessel retention of the molten core.
- The on-site management of consequences, especially of multi-unit accidents, has to be improved.

The opportunities for improvement of the accident management capabilities, identified based on the reviews performed in the framework of the stress tests, will need a timeframe for implementation. The HAEA has requested the licensee to establish a plan for implementation of the identified improvements, including completion dates. The plan has to be submitted by the 30th of June 2012.

5 List of acronyms

AC	Alternating Current
BCC	Back up Command Centre
BDB	Beyond Design Basis
Bf	Elevation above Baltic Sea level
CDF	Core Damage Frequency
CR	Control Room
DBE	Design Basis Earthquake
DBF	Design Basis Flood
EDG	Emergency Diesel Generator
ENSREG	European Nuclear Safety Regulator Group
EOP	Emergency Operating Procedure
ERO	Emergency Response Organization
ESWS	Essential Service Water
FSAR	Final Safety Analysis Report
HAEA	Hungarian Atomic Energy Authority
HCLPF	High Confidence Low Probability Failure
HNR	Hungarian National Report
IAEA	International Atomic Energy Agency
LOOP	Loss Of Offsite Power
MCP	Main Coolant Pump
NPP	Nuclear Power Plant
NSC	Nuclear Safety Code
PCC	Protected Command Centre
PGA	Peak Ground Acceleration
PSA	Probabilistic safety Analysis
PSHA	Probabilistic Seismic Hazard Assessment
PSR	Periodic Safety Review
RPS	Reactor Protection System
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SFP	Spent Fuel Pool
SG	Steam Generator
SHA	Seismic Hazard Assessment
SL	Seismic Level
SSCs	Structures, Systems and Components
TSC	Technical Support Centre
TSR	Targeted Safety Re-assessment
UHS	Ultimate Heat Sink
WWER	Vodo-Vodyanoi Energetichesky Reactor (= water moderated, water cooled energetic reactor)
WENRA	Western European Nuclear Safety Regulators' Association