Convention on Nuclear Safety
Questions Posted To Hungary in 2005

Seq. No	Country	Article	Ref. in National Report
1	Bulgaria	General	

Question/ Comment	What is the focus of the policy and the activity of the Hungarian Regulatory body and of the Paks NPP management for the period till the next national report review? What are the areas, improvements and plans for their implementation?
Answer	Focal areas of activity of the Hungarian nuclear safety regulatory body as well as of the Paks power plant for the next three years are: • elimination of the consequences of the April 2003 serious incident at Unit No. 2 • revision of the Act on Atomic Energy • power increase of all three units by cca. 8% • preparations to the life-time extension of the Paks NPP • preparing a new revision of the Nuclear Safety Rgulation series

Seq. No	Country	Article	Ref. in National Report
2	France	General	p. 1

Question/ Comment	The reports reviewed by France in view of the third peer-review meeting were all examined according to a standard list of issues derived from the obligations of the Convention. If an issue appeared to be covered in an incomplete way by the report of a Contracting Party, this led to a question or comment. However France recognizes that the corresponding information may be available in other existing documents.
Answer	The 3rd National Report of Hungary follows the structure of the first two reports. References to Articles of the convention are given in a cross-reference Table on page 7, and declarations (printed in italics) on pages 5, 17, 21, 27, 37, 39, 44, 48, 57, 63 and 71 further help to find the correspondence.

Seq. No	Country	Article	Ref. in National Report
3	France	General	table p. 7

Question/ Comment	The plan of the Hungarian report, to the contrary of most others, corresponds only very roughly to the suggested plan in the guidelines for national reports (INFCIRC/572/Rev. 2) and the various chapter in the body of the report never recall the articles of the convention they are dealing with: that makes it quite difficult to assess the report following appropriately the wording of the obligations of the Convention.
Answer	

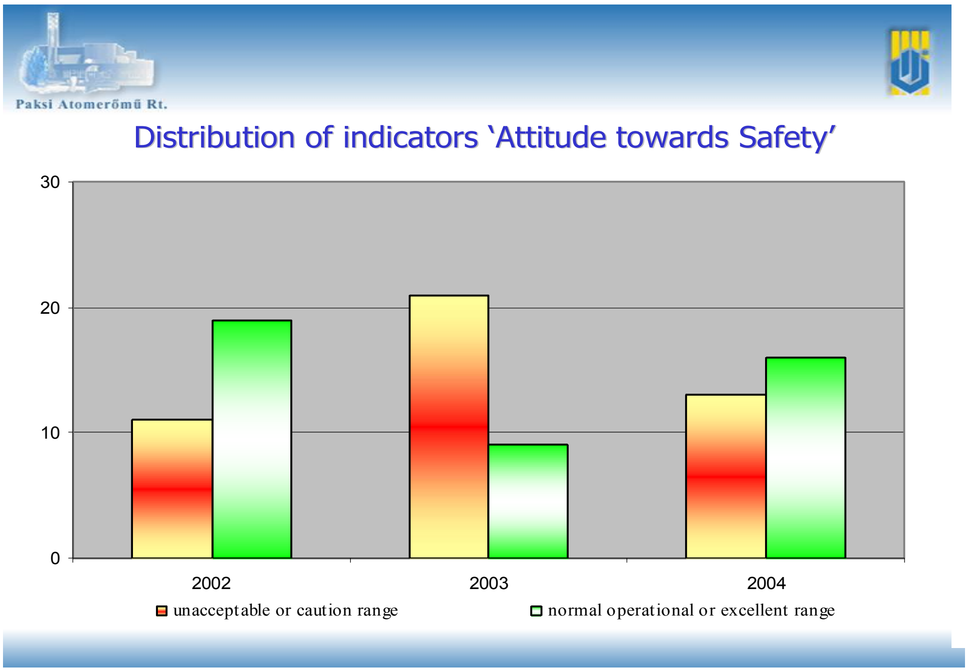
Seq. No	Country	Article	Ref. in National Report
4	Netherlands	General	Annex 7, page 109

Question/ Comment	In Annex 7 on page 109 it is stated: "Evolution of the above technical conditions leading to the incident was enhanced by several human factors. The most important of these are: in recent years at the plant, a number of small signs the deterioration of safety culture could be observed; the production was outstanding, but production interests gradually prevailed over safety. The management of the plant and the licensing authority placed too much reliance on the high prestige contractor." The incident happened in April 2003. Since then a number of technical and organizational measures were taken, especially with respect to improving safety culture. Is there a monitoring program to evaluate the development of safety culture? Are there any signs of improvement already?
Answer	

The Paks NPP has made a decision on review of former methodology of monitoring safety culture. The new monitoring program has the following three components:

- regularly evaluate the safety culture level on the basis of 'safety culture indicators' of Safety Performance Indicators System (SPIS)
- regularly (every year except when an international IAEA or WANO review is scheduled) assess the safety culture performing self assessment review in accordance with the relevant procedure ELJ-BIZT-05-04
- regularly perform the safety culture survey using opinion poll with questionnaire. The scope of the surveys should cover both employees and managers of the company and should also include those of the strategic partners of the company.

At the end of 2004 we analyzed the distribution of own 'Attitude toward Safety' indicators (practically these are the safety culture indicators of SPIS). In the Paks NPP SPIS the indicator value may be in four ranges: excellent (green), normal operational (white), caution (yellow), unacceptable (red).



This figure shows some positive signs of improvement in safety culture indicators in 2004 in relation to 2003 when the INES-3 event occurred.

Seq. No 5	Country Austria	Article Article 6	Ref. in National Report
Question/ Comment	What are the technical regulatory requirements for a lifetime extension of the NPP's?		
Answer	The legal framework of regulating nuclear safety relevant for lifetime extension is as follows: Atomic Act (Act CXVI of 1996) and the related 108/1997 Korm. Governmental Decree. They address the issue of lifetime extension as described below: • According to the atomic act a license (among them the operating license) may be granted for a defined or undefined period of time, as well as subject to certain stipulations. The license granted for a defined period may be extended when so requested. • The Governmental Decree issued for the execution of the act clarifies that the issuance of operating license could mean the extension of the designed lifetime.		

• According to the decree, in order to extend the design lifetime of the NPP units, not later than four years before the expiration of design lifetime, the Licensee shall submit a program to the regulator, which schedules the establishment of the conditions of the operability beyond the designed lifetime. The regulator inspects the program and its implementation.
• Licensing of operation beyond the design lifetime takes place through the new operating license issued before the end of design lifetime upon the application of the Licensee. Within the procedure assessing the application the regulator considers the results of the program and its inspection findings.

Detailed regulations
Within the Hungarian nuclear regulation system the detailed prescriptions are given in the Nuclear Safety Regulations. The regulations were issued as the appendices of the above mentioned governmental decree. There are six volumes of these regulations from which the first four is related to the NPP (the other two address the research reactors and spent fuel storage facility). This four volume divide the nuclear requirements as follows:
Volume 1: Regulatory procedures,
Volume 2: Quality assurance,
Volume 3: Design
Volume 4: Operation
Accordingly the regulations of different issues (for example of lifetime extension) are addressed by more than one volume.

The regulation divides the lifetime extension procedure into two stages:
a) program for lifetime extension,
b) new operating license.
a) Program for lifetime extension
According to the regulation the safe operation shall be continuously maintained during the preparatory phase and during the operation beyond the designed lifetime (OBDL) in accordance with the laws and regulatory prescriptions of legal force. The problems arising from the actual operation shall be handled within the valid operating license. During the OBDL the necessary safety margins, considered by the safety analysis, shall never be consumed, not even with reference to the approaching of the end of licensed lifetime. The activity aiming at maintaining the technical conditions of the safety SSCs shall be launched and continuously performed already within the designed lifetime; additionally the efficiency of this activity shall be systematically supervised and evaluated. The determination of safety improving measures, deriving from the modern international requirements, shall be carried out within the framework of PSR and not for the lifetime extension issue.

Requirements for the program aiming at establishing the conditions for lifetime extension:
• For establishing the conditions of lifetime extension and for the justification of operability the Licensee shall prepare a program. The program and a description of its time-proportional implementation shall be submitted to the regulator not later than four years before expiration of the design life. The program can be submitted for one or more units of the same plant. In the substantiating documentation at least 20 years of operating experience shall be considered. The regulator inspects the program and its implementation (and checks for any discrepancy that could prevent licensing of lifetime extension).
• All modification and fixing activity shall be performed within the framework of the valid operating license and not in the program.
• The program shall be based on the requirements for the application of the new operating license. Here the fulfillment or status of fulfillment or the activity (with schedule) planned for

the fulfillment of that requirements should be demonstrated.
• The program shall contain the planned duration of OBDL.

b) Operating license (OL)
Licensing of lifetime extension is performed in the new OL, upon the application of the Licensee to be submitted 1 year before the expiration of the lifetime. Validity: until defined time period if all conditions are fulfilled. In the OL application it should be demonstrated that:
• appropriate scoping of SSCs necessary to safe OBDL is performed;
• relevant ageing mechanisms are addressed;
• the condition of relevant SSCs are surveyed, efficiency of the former ageing programs are evaluated, new ageing management aspects and requirements are elaborated;
• scope of time limited ageing analysis (TLAA) involved in lifetime extension is determined, former TLAAs are re-evaluated and their validity is checked;
• the FSAR is actualized;
• necessary modification of operating conditions and limits are surveyed and substantiated;
• relevant documents (operating limits and conditions, maintenance policy, symptom-based emergency operating procedures, other emergency procedures, emergency response plan) are surveyed and their modifications necessary to lifetime extension are justified.
• Upon the above activities it is ensured that during extended lifetime the safety function are fulfilled at the desired reliability, the safety analysis covers the possible operating modes and the operating limits and conditions are in harmony with lifetime extension requirements.

The followings shall be attached to the OL application: actualized FSAR, modified version of the above documents, the necessary special authority contributions. Background documentation to the substantiating documents shall be submitted upon further regulatory request.

Re-licensing of operating and other licenses expired at the end of lifetime

Conditions for the issuance of the operating license: the temporary storage or final disposal of radioactive wastes and spent fuel shall be ensured in harmony with the international expectations and experience. The valid operating license is precondition; maximal length is the operating license of the unit. In the application the followings shall be demonstrated:
• The operation is in accordance with the approved safety analysis.
• The inspection, manual and emergency documents and procedures are appropriate for safe operation.
• Necessary initial data for condition monitoring of the SSCs are available.
• Safe operation is ensured fulfilling the operating limits and conditions.
• Technical and administrative conditions are ensured for long term safe operation, the financial resources performing long term maintenance and development of safety are available, the possible reasons for cancellation of the license are eliminated.
• The documents and contributions needed to OL are also parts of this application.

Guidelines relevant to lifetime extension
Besides the legally binding requirements the regulator has the possibility to issue not legally binding requirements. However these regulatory guidelines has important role in the system of regulations, because if the Licensee would like to deviate from the given guideline than it shall be justified that the applied method is more or at least equally conservative than the one of the guideline. This method shall be well substantiated.

The system of guidelines follows the structure of the Nuclear Safety Regulations; that is all of them are attached to one of the volume of the NSRs. So for example concerning ageing there are 4 guidelines in which four different aspects of requirements are address. In the guidelines the requirements of the NSRs are explained in details or the method of meeting the given requirement is formulated.

Concerning lifetime extension the following guidelines have already been issued:

Maintenance
1.19 Inspection of the efficiency of the maintenance program of the nuclear power plant
4.6 Nuclear power plant maintenance program and maintenance efficiency monitoring
Ageing
1.26 Regulatory Inspection of the Ageing Management Program
2.15 Quality Assurance in the Ageing Management of Nuclear Power Plant Equipment
3.13 Consideration of Ageing during Nuclear Power Plant Design
4.12 Management of Ageing During Operation of Nuclear Power Plants
Equipment qualification
1.27 Regulatory control over equipment qualification and preservation of the qualified status
3.15 Equipment qualification requirements during the design of nuclear power plants
4.13 Equipment qualification requirement for operating nuclear power plants

Additionally the two most relevant guidelines are under issuance. These guidelines directly address the lifetime extension. The titles and numbers will be:
1.28 Requirements for the scope of the lifetime extension license application
4.16 Conditions of operation license renewal of nuclear installations

Seq. No 6	Country Austria	Article Article 6	Ref. in National Report
Question/ Comment	What were the reasons for the decision to grant the license for the interim storage facility for 50 years?		
Answer	The expected commissioning date of the final national repository of the spent fuels is 2047. This is the main reason to design and licence the ISFS at least for 50 years.		

Seq. No 7	Country Austria	Article Article 6	Ref. in National Report
Question/ Comment	What are the procedures to ensure the safety of the interim storage facility over 50 years?		
Answer	On one hand the licensing procedures that require carrying out periodic safety reviews and re-licensing. According to this the present operational license will expire in 2008. On the other hand the main systems, items and components of the facility were designed for 50 years of lifetime or longer. The OLCs and the aging management procedures and monitoring that have been started at the early lifetime of the facility provide additional control on the safe operation and re-licenseability.		

Seq. No 8	Country France	Article Article 6	Ref. in National Report p. 14
Question/ Comment	The ageing management performed by Hungary (See §3.7.3 p. 45) to maintain the safe operation of the power plant beyond the lifetime leads to a list of critical components (Annex 2, p. 90). The report states that their replacement would be a challenge either because of their unique characteristics or because of their cost. List of equipment concerned contains essentially		

	mechanical equipment and cables and connections of safety related systems. Could Hungary clarify whether I & C equipment is considered in the ageing management?
Answer	The scope of the aging management program contains only passive and long life components (passive are the ones that perform their intended function without moving parts or changes in configuration or characteristics, long life are the ones that have no qualified lifetime or other lifetime restriction shorter than 30 years). The active components (for example most of the I & C equipments) are not considered in the ageing management because they are handled in the framework of the environmental qualification and maintenance activities.

Seq. No	Country	Article	Ref. in National Report
9	Germany	Article 6	p. 14, 1.1.2

Question/ Comment	The plant safety was thoroughly reviewed within the AGNES project (1996) and during PSRs (1996 and 1999). Modernization measures were extrapolated mainly on the basis of these safety reviews. Some of them have been already implemented, some others are planned. Please explain whether today (5 -10 years later) another decision (also in parts) would be taken regarding the scope and the sequence of implementation of modernization measures. Please consider in your answer the ongoing operational experience of the Paks plants as a whole and of the already implemented measures and characterize the cost risk-reducing ratio.
Answer	The modernisation or “Safety Enhancement Measures” (SEMs) planned on the basis of the original AGNES project have already been implemented almost in full. The most important implemented SEMs to mention are • the implementation of the symptom based EOPs (elaborated by Westinghouse), • the enhancements related to earthquake resistance, • independent emergency primary cool down system , • implementation of fully digital reactor protection system with enhanced logic (based on Siemens SW and equipment), • part of primary-to-secondary (PRISE) leak handling. The second part of PRISE leak handling modifications is still pending (in-containment emergency blow-down from the SGs), it is scheduled to be completed before the end of 2007. During the course of the last 10 years several modifications were carried out on the originally planned SEMs, for example originally for PRISE handling the qualification of the secondary blow-down valve for two-phase flow was proposed, but later it was deemed not being feasible, therefore decision was made to implement a new blow-down system inside the containment. In compliance with the best international practice, a detailed Severe Accident Management Guideline is going to be developed, as a new SEM. The detailed schedule of its elaboration and implementation will be completed this year. The details of the planned SAMG are greatly based on the level 2 PSA studies, which were completed last year and were reviewed by an international expert team under financing of a PHARE project. Since it is related to the level 2 PSA, the cost/risk-reducing ratio plays an important role in the final decisions. CDF was reduced by more than an order of a magnitude due to the implemented SEMs, however, we are not aware of the cost of the implementation. The feed-back from operational experience plays an important role in the regular updating of the safety of the plant. An important example is the proposed extension of the symptom based EOPs for initiating events occurring at different shut-down states. This decision is not only initiated by the revealed high contribution of shut-down states to Early Large Release by the level 2 PSA, but also some operational incidents (including the fuel cleaning incident last year) emphasized its significance.

Seq. No 10	Country Germany	Article Article 6	Ref. in National Report p. 14, 1.1.2

Question/ Comment	How is it insured that the procedures are updated in compliance with the hardware modifications? Are there obligatory procedures for periodical verification of this compliance?
Answer	During the procedure of the licensing in principle the Licensee has to present a list of plant documentations affected by the given plant modification and there it has to give a qualitative description of the changes in the documents. Some of the documents (e. g. Operational Limits and Conditions) may be modified only if it is approved by the HAEA NSD. This approval procedure is the part of the plant modification licensing procedure. Another documents shall be presented to HAEA NSD in case of a change. There is no procedure for periodical verification of compliance. HAEA NSD checks the operational documents within the framework of its normal inspection programme.

Seq. No 11	Country Germany	Article Article 6	Ref. in National Report p. 14, 1.1.2

Question/ Comment

This question also relates to Article 14: see page 45, chapter 3.7.3 of your Report
With regard to §3.7.3 PLIM is performed according to the regulations. Does this PLIM consider possible PLEX? Please provide some examples about the first results and the experience feedback.

Answer

Yes, because the NPP performed a feasibility study and as the result of it one of the strategic goals of the company is to extend the service life of the four units of the NPP by 20 years beyond the design lifetime.
Extension of the operational lifetime is a strategic decision that is entirely based on the design- and manufacture features of main components the WWER/440/213 type units at Paks; on the system of technical inspections and tests; the maintenance practice; as well as on the good condition of the plant maintained via reconstruction and refurbishment during last years.

Example 1: Reactor Pressure Vessels
As for the reactor vessels of WWER/440/213 at Paks NPP the dominant ageing process is caused by embrittlement due to fast neutron irradiation of the material. The vessels are different per units and their lifetime can be extended under different conditions. At units 3-4 reactor vessels do not require extra measures even at 50 years of lifetime. For the lifetime extension at unit 2 the heating up of the water in the emergency core cooling (ECC) tanks may be considered in order to decrease stress levels caused by pressurized thermal shock (PTS) transients. For this purpose cost-effective technical solutions are available. At unit 1, in case of the 50-year lifetime, in addition to the potential ECC heating-up, the annealing of the welded joint No. 5/6 close to the core will be considered.
Although the vessel walls and the next to the core circumferential welds have good embrittlement properties, because of their relatively low contaminant material quantity (Manufacturer SKODA), low leakage core configurations are used according to the life time management program, and like this the fast neutron fluence could be decreased with more than 30 percents.
The significantly decreased irradiation load not only makes a good base for the operational life time extension for 50 years of the vessels, but allows of the reactors' power up rating (~8%) without further impairment of the vessels.

Example 2: Steam Generators
The technical condition of the 24 horizontal steam generators belonging to the four units at Paks, play serious part in the units' economically competitive life time extension. In the case of the steam generators' austenitic heat-exchange tubes the lifetime limiting ageing mechanism is

the Outer Diameter Stress Corrosion Cracking. Considering the operation experience of all VVER power plants PWSCC like failures of the tubes have never been detected. For assuring integrity of the steam generator tubes 100% Eddy-current inspection is performed. More then 80% of the indications are originating from locations of tube supports plates, where the secondary circle corrosion products with concentrated corrosive agents are easily to be absorbed. This phenomena can be significantly mitigated by decreasing corrosion product level in the secondary circle.
The reserve in heat exchange area of the VVER steam generators is more then 15% according to the designers preliminary analysis. Considering the recent plugging trend of the heat exchange tubes constant, expectedly none of the steam generators would exceed the 15% plugging rate (<10%) by the end of its 50 year life time. Also – despite of the fact that not all types of circumstances affecting ODSCC are known – it is known for sure, that by the known value of some basic parameters better plugging trend can be expected, also corroborated by the secondary circuit water chemistry modifications and the selection of new structural materials (copper removal, austenitic condenser tubes, erosion-corrosion/erosion resistant high pressure pre-heater tubes, other erosion proof modifications, increasing feed water pH to eliminate erosion/corrosion, minimizing corrosion activator level etc.).
We have reasonable chance that the described ageing management steps will slow down ODSSC in the coming operating periods.
In summary, according to our present knowledge the steam generators at all four units of the VVER power plant at Paks can continue to operate during the lifetime extension without replacing any of them and also have reasonable reserve for the planned power uprating.

Example 3: Pressurizer and surge-line aging management
Aging management programs are specified for the dominant degradation mechanisms of all sites. In the case of the pressurizers fatigue is supposed to be the major life limiting degradation mechanisms for the location of water injection nozzle and the surge nozzle with the surge line.
To support the life extension option of the pressurizers site specific fatigue monitoring programs are used. In the case of the injection nozzle, partial cycle counting method, based on partial water injection temperature differences is used. Sample type fatigue monitoring has been used by processing surface temperature measurement results for AMP of the surge line.
Other possible degradation mechanisms are managed mainly by ISI programs conducted every 4 years.
By our present knowledge the pressurizers and surge lines at the four units of the VVER power plant in Paks can continue to operate during the life time extension without replacing any of them

Example 4: Other safety related equipment aging management
Aging management of safety related piping, vessels, heat exchangers, pump/valve casings is covered by so called technical condition control performed every 4/8 years. The typical technical condition control consists of pressure test, visual examination and sample type non-destructive tests in high safety significant systems.
Secondary circle equipment locations affected by erosion/corrosion are controlled under special wall thickness measurement programs based on erosion-corrosion AMP expert system.
Systematic material replacement of these components under the umbrella of the life cycle management program continuously supports the life extension option of these components as well.

Seq. No	Country	Article	Ref. in National Report
12	Slovenia	Article 6	section 1.1.2, p 14

Question/ Comment	The third paragraph states that the conclusions arising from the (national and international) reviews were on the wholly positive. A long list of missions are presented in table 4.3.8-1. One root cause that lead to the incident in April 2003 was also maintenance of steam generators and their decontamination what lead to deposits on fuel. What were the recommendations of these missions concerning this problem? Explain the statement that conclusions of this large number of missions were on the wholly positive and the relation to the 2003 incident. IAEA mission reviewing the incident pointed out weak points – what is the implementation status of these recommendations?
Answer	Those missions did not discuss those long-term technical problems or if there were any discussions during those missions no related recommendations were made. It seemed that the problems were treated properly and only very deep analysis could have revealed that the treatment of the problem had not been correct. International missions are not suitable for such deep technical analysis. The problem was that the plant did not properly take into account the recommendations made by the Hungarian Nuclear Safety Authority regarding those long-term problems such as steam generator feed-water header replacement and deposits on the fuel assemblies. Regarding the "positive" result of the previous missions (concerning those conducted before the year of 2000) the situation is the following. The general conclusions of those missions were positive. All missions, however, had identified several problems which were treated by the plant in the action plans prepared after the respective missions. Some of the problem areas reappeared in the investigation- and mission-reports after the 2003 incident (e.g. deficiencies in the implementation of operating experience). However, most of the identified root causes of the Paks 2003 incident were not identified as problem areas in the previous mission-reports. Regarding the IAEA incident review mission, the plant has prepared a so-called Comprehensive Action Plan (CAP). Actually the actions in the CAP were initiated by the plant's own investigation and by the investigations performed by the Hungarian nuclear safety authority and the IAEA review mission. The Action Plan is in the phase of implementation. This means that most of the short-term actions have already been completed. There are a number of actions with long-term effects particularly those dealing with safety management and safety culture improvement. This type of actions are in the implementation phase and some of them are planned to be completed by the year 2006. An IAEA mission is conducted at Paks in February 2005. The objective of the mission is to review the progress made by the plant since the OSART mission in 2001 and since the expert mission conducted in June 2003 after the Unit 2 incident (in fact in order to review the status of the implementation of CAP).

Seq. No	Country	Article	Ref. in National Report
13	Austria	Article 7	

Question/ Comment	What are the procedures to guarantee state-of-the-art regulations?
Answer	The HAEA Nuclear Safety Directorate has the duty to participate in the revision of the Nuclear Safety Regulations to be performed once in every 5 years, and it is also a partner in drafting of new regulations. The HAEA NSD is in charge to elaborate, to harmonise with other organisations, to finalise and to make the necessary technical steps to issue Safety Guides on fulfilment of nuclear safety regulations. The HAEA NSD also participates in drafting of new legal documents having relevance to its authority duties, and in the administrative harmonisation procedure of bills having relevance to

its duties.

HAEA observes periodically the new IAEA guideline developments, while its representatives are participating in the activities of Nuclear Safety Standard bodies of the IAEA. HAEA is informed about the existing and changing EU directives.

The HAEA NSD is participating in the WENRA activities in order to contribute to the regulatory requirement harmonisations in the fields of nuclear safety and waste management.

The above activities are supported with several written procedures for the HAEA staff.

In conclusion there are sufficient information sources available for the HAEA NSD in order to learn about state-of-the-art requirements, common approaches and good practices.

Seq. No 14	Country Austria	Article Article 7	Ref. in National Report
Question/ Comment	Which consequences have been taken from the incident on 10 April 2003 with regard to licensing procedures ?		
Answer	After the incident a revision of the licensing activity at HAEA NSD has been initated. A temporary working group reviewed the Regulations, Guides and Procedures related to the licensing activity. The group concluded that two of the most relevant documents (one guide and one procedure of HAEA NSD) should be united and the resulting procedure should be completed with more detailed guidance for the personnel of HAEA NSD. The most important development is that the responsible inspector of a given licensing procedure shall follow and fill in a table containing all the requirements related to the type of the actual licensing procedure. In the table it shall be confirmed that the licensing documentation has been checked against the respective requirements one-by-one. The various licensing procedures and regulations are directly linked with the procedure. Another important consequence of the revision is that there is more stress on the team work: in the case of a complex licensing documentation the responsible inspector shall involve the necessary experts into the work and the rules of this process are laid down in the Procedure. The Nuclear Safety Regulations have been reviewed and those IAEA SSS recommendations and requirements that have not been addressed before were also taken into account. The most important element of this was the inclusion of the safety categorisation of the plant modifications into the licensing requirements. The Nuclear Safety Regulations are due to enter into force in mid 2005. After the incident the licensing activity has also been affected by a re-organisation of the HAEA NSD. On one hand a so-called vertical system is introduced, where the responsible inspector is involved all along the licensing process (preliminary discussions, licensing activity, related inspections etc.). On the other hand, licensing pertaining to the various facilities are separated, thus a separate division is dealing with the tasks related to the Paks NPP. It is expected that as the result of the above changes the personal responsibility will be more definite and the evaluation process of the documentations becomes more uniform (independent from the person who compiles it) than before.		

Seq. No 15	Country Austria	Article Article 7	Ref. in National Report

Question/ Comment	The CNS report states, "In addition to unambiguously recognizing the severity and significance of its consequences it is also justifiable to point out that the event did not take place in the technological systems necessary for the normal service of the plant, but it occurred inside a cleaning tank designed and operated by an outside contractor, during the shutdown state of Unit 2. Thus the lessons learnt from the event have in no way changed the evaluation of the nuclear safety of the plant, as a technological installation." In this regard: (a) What is the reason for the suggestion that refuelling is not part of the "normal service" of the plant, and that as a consequence systems and structures used in refuelling are not part of the technological systems necessary for normal service of the plant?; (b) What is the relevance, according to the provisions of the Convention on Nuclear Safety, of the statement that the Paks Unit 2 incident "occurred inside a cleaning tank designed and operated by an outside contractor "?; and (c) What is the relevance, under the provisions of the Convention on Nuclear Safety, to the occurrence of the Paks Unit 2 fuel damage incident while the reactor was in a shut down state?
Answer	(a) The incident occurred in the cleaning tank had not effected the normal operational systems and components of the reactor. The consequences and influence of the incident was limited to service shaft no. 1. Shaft no. 1 was separated from the spent fuel pool and the refuelling was accomplished without using the shaft. The refuelling is really part of the normal operational activities (the report does not state the contrary) but the fuel cleaning is not. (b) (c) The incident occurred in the cleaning tank has been the worst nuclear safety related issue in the history of the plant. Because of the international interest and the lessons from the incident we deemed important to include the causes, contributing factors and the corrective measures into the report.

Seq. No	Country	Article	Ref. in National Report
16	Austria	Article 7	

Question/ Comment	The entering into force of the revised regulations was impeded by the incident of 10 April, 2003. When will the revised safety regulations be implemented?
Answer	Revision of the Nuclear Safety Regulations has been restarted right after the incident in order to include recent recommendations by international organisations. The revisions have been completed in 2004, harmonisation with the licensees and with other governmental organs has been performed and the regulations are waiting for a governmental approval and the issuance of a governmental decree . They come to force after the promulgation of the Governmental Decree immediately.

Seq. No	Country	Article	Ref. in National Report
17	Canada	Article 7	2.1, page 18

Question/ Comment	The report indicates that effective 1 August 2003 the supervision of HAEA "was handed over by the Minister of Economy and Transport to the Minister of Interior." Please explain the rationale for this hand over.
Answer	According to the 1996 wording of Act on Atomic Energy: "The Government shall exercise supervision over HAEA through the President of the HAEC" (Hungarian Atomic Energy Commission). From 1994 to 1 August 2003 the Minister of Economy and Transport chaired the HAEC. By the amendments of Act on Atomic Energy in 2003, taken into force by 1 August, HAEC was abolished. In the new formulation the Act reads as: "It [HAEA] shall be supervised by a minister appointed by the Prime Minister". Meanwhile the review of the nuclear safety in Hungary carried out by experts of the European Union recommended the separation of the functions of promotion of energy production (one of the tasks of the Minister of Economy and Transport) from the surveillance of the safety

	authority. As a consequence of the legal change and the EU recommendation, by 1 August 2003 the Prime Minister appointed the Minister of Interior to supervise HAEA.		
Seq. No 18	Country Korea, Republic of	Article Article 7	Ref. in National Report
Question/ Comment	(Page 17, 2.1 The Act on Atomic Energy) It is stated that in your report some ministries and central administration bodies are respectively fulfilling regulatory tasks in the field of safe application of nuclear energy, nuclear safety, and radiation protection. 1. If so, in case of a Nuclear Power Plant, should the applicant obtain permits both from the HAEA with regard to the safety of the NPP and from the Ministry of Health with regard to radiation protection? 2. Please provide us the name of other important ministries and central administration related with the license of NPP together with their respective tasks.		
Answer	1. Indeed, this is the case. However, the basic licenses are issued by the HAEA NSD and the existence of the permission from the Authority for radiation protection is a pre-condition of the basic license. It only means a time sequence, there is no other type of hierarchy between radiological permissions and basic licenses, that is basic licenses must not change or overwrite any decision of their pre-conditional permissions. 2. Ministries delegate the right of authority to institutes or organisations in the area of responsibility assigned to the given ministry in the Atomic Energy Act. Some ministries and the areas of responsibilities are mentioned below. Ministry of Public Welfare: radiation protection Ministry of Environmental Protection: environmental protection nature conservation water quality protection Ministry of Interior: fire and physical protection emergency preparedness public and internal order Ministry of Economy: geology Ministry of Transport and Water Management water utilization water base protection etc. (regional planning and building, mining technology and safety, measuring instruments, food, plant and animal hygiene, soil protection)		
Seq. No 19	Country France	Article Article 7.2	Ref. in National Report p. 22
Question/ Comment	The report mentions that "a separate license shall be obtained for all plant level or safety related equipment level modifications". Does the regulator have the capability to assess and review in details all the modifications normally occurring in the lifetime of an installation? Can this exempt the operator to conduct its own independent internal assessment before submitting an application for all these authorisations?		
Answer	It is to be underlined that – as it is stated in the text - only modifications related to safety are falling under licensing obligation. Another important issue, that the Hungarian safety regulations follow an elaborated in detail graded approach when prescribing what kind of		

licenses are required for different modifications. Accordingly for the less safety significant modifications only the basic ideas/principles are submitted to and approved by the Authority. For identical modifications of the equipment (e.g. in case of replacement of one type of a valve with another type) so called “standard” (or typical) licences may be issued. On the other hand in case of more complex modifications a multi-step licensing procedure is in place consisting of the licenses in principle, for fabrication, installation, commissioning and operation activities separately. If the categories are defined carefully, but with due consideration, the regulator may have the necessary capability to assess all the modification requests. Concentration of the efforts of the regulatory personnel to the questions having real safety significance is an ongoing activity for several years as part of drafting of amendments to the safety regulations.
The safety regulations clearly define, for which category of applications has to be attached an independent internal assessment, and which require attachment of an independent assessment from an outside expert. The principium is stipulated in the Act on atomic energy: the users are responsible for safety of application of atomic energy and compliance with safety requirements, therefore the operator can not be exempt from its responsibility by any action of the regulatory authority.

Seq. No	Country	Article	Ref. in National Report
20	United States of America	Article 7.2.1	Section 4.3.1, p.72
Question/ Comment	Approval is necessary for modification of the Final Safety Analysis Report. It is not clear what specific changes (other than the FSAR) require approval. What plant design, equipment and test changes may be made by the licensee, and which must receive regulatory approval prior to implementation? What documentation substantiates the approval?		

Answer

According to the regulations in force:

A modification means all such alterations, excluding the ones belonging to the concept of repair related to systems, system elements of the nuclear power plant’s one or more units which result in change of

a) systems and system elements, or
b) operational conditions and limits, or
c) requirements recorded in technical and administrative documents regulating operation (for example, codes, instructions, procedures), or
d) requirements specified in a safety report,

independently of that the unit or units is (are) under construction or commissioning, or already in operation. Also alteration of the organization providing the safe operation and its conditions, as well as alteration of the management shall be considered as a modification.

Modifications can be classified into three categories. Classification into modification categories shall be based on the preliminary assessment and have the following particularities.

The classification is identical to that recommended by IAEA SSS NS-G-2.3

In case of modifications classified into the 1st and 2nd modification categories, this approval is executed during a multistage process which, depending on the complexity of the modification, may result in issuance of the following licence types:

a) modification licence in principle,
b) production, procurement licence of system elements related to the modification,

c) installation licence of system elements related to the modification,
d) modification licence,
e) operational licence.

In case of modification classified into the 3rd modification category in order to approve the categorization, the preliminary assessment of the modification purposed and documents serving as the base of it shall be submitted to the Authority.

Content requirement of the application for modification licence
In the application for licence, it shall be certified that safety of the unit does not decrease even during the construction phase, under the modified system or systems and system elements it is possible the unit to be safely taken in service and operated. In order to provide this, the application for licence shall contain:

a) description of demolition and installation activities needed for implementation of the modification
b) a certification of that the state implemented on the basis of activities specified in item a), as well as of licences earlier issued for implementation of the modification (production, procurement and installation licence) or of works executable without authorial licence harmonizes with requirements specified for the modification, system elements concerned in the modification, their installation place and their safety class in Nuclear Safety Regulations, other legally binding statutes, the safety review report of the facility and in the documentation confirming the modification,
c) the summary of deviations from the documentation confirming the modification licence in principle, and confirmation of these deviations,
d) the quality assurance programme demanded to be implemented during construction of the modification,
e) the programme of trials, commissioning activity needed for certification of scheduled operation of the modified part, a comparison between the programme and the requirements made against the commissioning and trials in the licence in principle, acceptance criteria of the successful trial and commissioning and their confirmation, as well as personal and documentational conditions of the trial, commissioning,
f) such changes of operational conditions and limits of the nuclear power plant, the condition-oriented emergency-prevention instruction, the emergency preparedness action plan and in accident-management processes which occurred due to the alteration and are needed in order to safely operate the unit concerned in the modification, and confirmation of these changes,
g) description and confirmation of frequency, scope and method of in-service investigations, inspections to be performed in order to provide availability of function of the modified system, system element,
h) presentation of alterations to be performed in instruction and maintenance books needed for operation due to the modification, specifying the list of instructions concerned and schedule of their completion, as well as a summarizing description of the necessary alterations,
i) the draft alteration, due to the modification, of requirements specified in the Final Safety Review Report, which is demanded to be introduced at annual actualization of the Final Safety Review Report, unless otherwise ordered by the Authority in the modification licence in principle
j) the training programme needed due to the modification, as well as presentation of conformity of the programme (subjects educated, the scope of people concerned in the education, schedule, the inspection method for fulfilment of the educational purposes),
k) licence(s) issued by the Authority in relation to the process, as well as the name and

identification numbers of the documentation earlier submitted by the Licensee, used for confirmation of the application,
l) approvals on the part of professional authorities in accordance with the legally binding statutes.

Seq. No 21	Country United States of America	Article Article 7.2.3	Ref. in National Report Section 2.2.3

Question/ Comment

Section 2.2.3 states that "the risk informed approach remained conservative. The number of probabilistic goal values and parameters have not grown, neither have their values changed." Does the Authority have plans for expanding its risk-informed regulation? If so, what is the schedule for developing these techniques?

Answer

Yes, it is planned to follow the trend to a risk-informed approach. For this purpose an Implementation Plan was prepared and approved at the HAEA and a comprehensive long term project was launched in 2003. The Risk-informed Implementation Project (RIP) schedules all the tasks, which have been identified important to improve the legislative, modelling and training areas as prerequisites for the successful implementation. The phases are shown in the attached table.

	A) Legal Background	**B) Models and tools**	**C) Expertise**
1	Policy on RIDEM	Requirements for PSA models, tools and studies	Training of inspectors on available PSA models
2	Graded QA (SSC, processes)	Review of PSA models	Training on application of PSA tools
3	Elaboration of probabilistic criteria	Development of tools for RIDEM	Trial application of RIDEM tools and procedures
4	Risk-informing the NSC and Guides		
5	Risk-informing the Nuclear Safety Guides		
6	Establishment of RIDEM procedures		

(yellow)	Tasks in Phase I, Establishment: 2003-2006
(blue)	Tasks in Phase II, Implementation: 2007-2008
(red)	Tasks in Phase III, Introduction: 2009-

Seq. No 22	Country Austria	Article Article 8	Ref. in National Report

Question/ Comment

What is the status of the changes at the Hungarian Atomic Energy Authority according to two recommendations and four suggestions of the IAEA IRRT mission?

Answer

Recommendations

R.1. The roles and responsibilities of all parties/authorities involved in the Hungarian nuclear regulatory process should be clarified and formalized in more depth. Additional administrative and/or legal actions seem to be necessary in order to consolidate the relationship of parties

concerned.
Status: legal study has been performed, which induce some changes in the Atomic Energy Act, in Governmental Decree describing the activities of the HAEA, and in Ministerial Decrees about Health, Environment Protection, Building and Fire Safety Authority activities. Several negotiations have already been performed in order to prepare the legal changes. Among the changes the new Act on the general state authority rules has the priority. The Act has been promulgated in December 2004, and all other changes should be harmonized with it.
R.2. In order to achieve comprehensive independence of the HAEA a governmental action is necessary to transfer the responsibilities/rights of the DG of the HAEA as the Founder of PURAM as well as the HAEA as Manager of the Central Nuclear Financial Fund to another governmental body.
Status: The issue will be handled in the modifications of the Atomic Energy Act, but after the harmonisation with the new Act on the general state authority rules. Some elements of the Central Nuclear Financial Fund's mission are discussed on a political level in the recent months.

Suggestions

S.1. NSD should continue in its discussions with Government to preserve the situation with regard to the current time limits associated with license applications.
Status: During the harmonisation with the new Act on the general state authority rules, the Atomic Energy Act modifications establish new safety authority time limits.
S.2. The HAEA should define its support role as an independent and credible spokesperson with respect to public information in its own Emergency Plan and procedures including staff support, and should continue its efforts to clearly define its role in the National Emergency Plans and procedures under development.
Status: the HAEA and the National Emergency Plan has been completed, with satisfactory result from this aspect.
S.3. The HAEA is encouraged to complete the agreement on cooperation between the HAEA and the State Public Health and Medical Officers' Service.
Status: The agreement has been signed, and today it is planned to extend to a three party agreement, with the involvement of the Environment Protection Authorities.
S.4. In the inspection plan on TRP a focused inspection should be included on the radiation monitoring system (equipment, methods, procedures) for balancing of releases into environment. This inspection should preferably be a joined co-authorities inspection.
Status: Co-authority collaboration has been exercised in the cases of the NPP's Emission Code, and regarding the Environment Monitoring System refurbishment. Occasional checking of environment monitoring data took place during and after the 2003 fuel event. The data acquisition had been found reliable.

Seq. No	Country	Article	Ref. in National Report
23	Austria	Article 8	
Question/ Comment	What is the trend in regulatory staffing in the past three years, and how is it planned to maintain sufficient regulatory staffing in the future?		
Answer	The number of staff of the regulatory body has changed two times in the last three years. In 2003, due to a governmental level civil servant staff reduction the staff was decreased by three persons. In view of the special role and position of the Hungarian nuclear safety authority five new positions have been opened in 2004. For the limited financial possibilities these positions could have been filled in only by recently graduated, inexperienced engineers. An intensive and concentrated training program shall bring the new inspectors to an adequate professional level		

	in a 1,5 years period of time. In view of the tasks foreseen in the near future further increase of the staff with 2-3 persons would be desirable.		
Seq. No 24	Country Austria	Article Article 8	Ref. in National Report
Question/ Comment	Which individuals and organizations constitute the Technical Support Organisations (TSOs) of the HAEA?		
Answer	The TSOs of the HAEA are: • the KFKI Atomic Energy Research Institute, • the VEIKI Electric Energy Research Institute Co. , • the Nuclear Technology Institute of the Budapest University of Technology and Economy, • the Physical-Chemistry Institute of the Veszprém University, • the Frederic Joliot Curie Institute of Radiology and Radio-Hygiene, • the Isotopic and Surface Chemistry Institute, which institutions – by bilateral agreements – are available at short notice in case of necessity of urgent advising and support in nuclear safety issues. These institutions regularly provide technical support to HAEA also on a contractual basis.		
Seq. No 25	Country Austria	Article Article 8	Ref. in National Report
Question/ Comment	In chapter 2.2 of the report it is described that the supervision of the Hungarian Atomic Energy Authority (HAEA) changed from the Ministry of Economy and Transport to the Ministry of Interior on 1 August 2003. The report states that the activity of the HAEA was surveyed for the second time by an IAEA IRRT (International Regulatory Review Team) mission in 2003 which focused on the HAEA NSD. The mission stated that HAEA had made efforts regarding to the recommendations and suggestions of the mission in 2000. The mission formulated two recommendations and four suggestions. --- The HAEA is responsible for waste disposal and decommissioning, and also serves as the administrator of the Central Nuclear Financial Fund. Isn't looking after both these tasks a conflict of interest for the HAEA?		
Answer	The comment “the HAEA is responsible for waste disposal and decommissioning” is not precise, HAEA is not responsible for waste disposal and decommissioning. According to Act on Atomic Energy: “As the solution of such matters is in the national interest, the performance of tasks related to the final disposal of radioactive waste, as well as to the interim storage of spent fuel, and to the decommissioning of a nuclear facility shall be the responsibility of an organisation designated by the Government.” The designated organisation is the Public Agency for Radioactive Waste Management (see p. 81 of the National Report).		
Seq. No 26	Country Bulgaria	Article Article 8	Ref. in National Report page 20
Question/ Comment	The report presents the information that in the most of the areas of the IAEA IRRT recommendations a significant step ahead was made. Can Hungary provide examples when this was not the case and what measures were introduced additionally?		
Answer	Recommendation-4 (2000) The legal and governmental infrastructure of Hungary with distributed regulatory responsibilities, involving up to nine authorities, should be more thoroughly co-ordinated in order to avoid any omission or overlap and to provide for effective co operation between those authorities.		

Comment: The complex authority structure hasn't changed despite of existing arguments about centralisation especially on the field of nuclear power. As a partial remedy, Governmental Co-ordination Council on Atomic Energy had been established in 2004.

Recommendation-7 (2000)
The time period in which the resolutions of the HAEA-NSD are to be delivered should not be constrained to avoid compromising the regulatory body's responsibility.
Comment: The time available for regulatory resolutions is determined by the general state authority rules. As an exception, the Atomic Energy Act allowed 6 months for installation level licensing. We had the first chance to propose extended time periods for other safety significant licensing case types by the end of 2004, when new act about state authority rules had been elaborated.

Suggestion – 5 (2000)
The need for reviews of resource allocation should be performed periodically having in mind that regulating and supervising the use of nuclear energy effectively, is an evolutionary process rather than a static one. The periodic review should also consider potential needs of salary adjustments. Incomes of regulatory staff should be comparable for the equivalent positions of counterparts within the nuclear industry.
Comment: The incomes of the regulatory staff remained under the control of Act about the status of Civil Servants.

Seq. No 27	Country Bulgaria	Article Article 8	Ref. in National Report page 25
Question/ Comment	The report presents that coming into force of the revised regulations was impeded by the serious incident at Paks NPP in April 2003. Could Hungary explain in more details what exactly was deemed additionally necessary?		
Answer	The April 2003 fuel event HAEA and IAEA investigation revealed the miss-interpretation of Safety Classification during the fuel assembly cleaning system design and licensing, and the IAEA recommended to utilize various elements of the newest safety standard series, issued between 2001-2002. The HAEA NSD restarted the review of the safety regulations, and the staff went through all the relevant standards and has identified the principles missing from the Hungarian regulations. Among other issues this has lead to the new system of safety categorisation of plant, system and equipment modification initiatives.		
Seq. No 28	Country Canada	Article Article 8	Ref. in National Report 2.2.1, page 22
Question/ Comment	The report states that "the safety-informed licensing of a nuclear installation takes place after the environmental protection licensing". Please explain what is meant by the term "safety-informed licensing".		
Answer	"The safety-informed licensing of a nuclear installation" phrase is really an example of the changes a text may undergo when it is translated. In this case the original Hungarian text mentioned "licensing from the nuclear safety point of view", or simply the licensing as considered in Article 7., paragraph 2. ii. in the Convention on Nuclear Safety.		
Seq. No 29	Country Canada	Article Article 8	Ref. in National Report 2.2.1, page 23
Question/	What are the fixed periods for which the units of Paks are licensed?		

Comment What are the regulatory requirements that the licensee has to satisfy to be granted an extension to the license?
How do the results and the timing of a Periodic Safety Review influence the granting of an extension of a license?

Answer 1. The designed lifetime of the units is 30 years. The operating licenses of units No. 1 and 2 are valid until 2008, those of units No. 3 and 4 until 2010 when the units shall undergo periodic safety reviews.
2. The legal framework of regulating nuclear safety relevant for lifetime extension is as follows: Atomic Act (Act CXVI of 1996) and the related 108/1997 Korm. Governmental Decree. They address the issue of lifetime extension as described below:
• According to the atomic act a licence (among them the operating licence) may be granted for a defined or undefined period of time, as well as subject to certain stipulations. The licence granted for a defined period may be extended when so requested.
• The Governmental Decree issued for the execution of the act clarifies that the issuance of operating license could mean the extension of the designed lifetime.
• According to the decree, in order to extend the design lifetime of the NPP units, not later than four years before the expiration of design lifetime, the Licensee shall, submit a program to the regulator, which schedules the establishment of the conditions of the operability beyond the designed lifetime. The regulator inspects the program and its implementation.
• Licensing of operation beyond the design lifetime takes place through the new operating license issued before the end of design lifetime upon the application of the Licensee. Within the procedure assessing the application the regulator considers the results of the program and its inspection findings.

Detailed regulations
Within the Hungarian nuclear regulation system the detailed prescriptions are involved into the Nuclear Safety Regulations. The regulations were issued as the appendices of the afore mentioned governmental decree. There are six volumes of these regulations from which the first four is related to the NPP (the other two address the research reactors and spent fuel storage facility). This four volume divide the nuclear requirements as follows:
Volume 1: Regulatory procedures,
Volume 2: Quality assurance,
Volume 3: Design
Volume 4: Operation
Originating from this fact it concludes that the regulation of different issues (for example lifetime extension) is addressed by more than one volume.

The regulation divides the lifetime extension procedure into two stages:
- program for lifetime extension,
- new operating license.
a) Program for lifetime extension
According to the regulation the safe operation shall be continuously maintained during the preparatory phase and during the operation beyond the designed lifetime (OBDL) in accordance with the laws and regulatory prescriptions of legal force. The problems arising from the actual operation shall be handled within the valid operating license. During the OBDL the necessary safety margins, considered by the safety analysis, shall never be consumed, not even with reference to the approaching of the end of licensed lifetime. The activity aiming at maintaining the technical conditions of the safety SSCs shall be launched and continuously performed already within the designed lifetime; additionally the efficiency of this activity shall be systematically supervised and evaluated. The determination of safety improving measures,

deriving from the modern international requirements, shall be carried out within the frame of PSR and not for the lifetime extension issue.

Requirements for the program aiming at establishing the conditions for lifetime extension:
• For establishing the conditions of lifetime extension and for the justification of operability the Licensee shall prepare a program. The program and a description of its time-proportional implementation shall be submitted to the regulator no later than four years before expiration of the design life. The program can be submitted for one or more units of the same plant. In the substantiating documentation at least 20 years of operating experience shall be considered. The regulator inspects the program and its implementation (and checks for any discrepancy that could prevent licensing of lifetime extension).
• All modification and fixing activity shall be performed within the frame of the valid operating license and not in the program.
• The program shall be based on the requirements for the application of the new operating license. Here the fulfilment or status of fulfilment or the activity (with schedule) planned for the fulfilment of that requirements should be demonstrated.
• The program shall contain the planned duration of OBDL.

b) Operating license (OL)
Licensing of lifetime extension is performed in the new OL, upon the application of the Licensee to be submitted 1 year before the expiration of the lifetime. Validity: until defined time period if all conditions are fulfilled. In the OL application it should be demonstrated that:
• appropriate scoping of SSCs necessary to safe OBDL is performed;
• relevant ageing mechanisms are addressed;
• the condition of relevant SSCs are surveyed, efficiency of the former ageing programs are evaluated, new ageing management aspects and requirements are elaborated;
• scope of time limited ageing analysis (TLAA) involved in lifetime extension is determined, former TLAAs are re-evaluated and their validity is checked;
• the FSAR is actualized;
• necessary modification of operating conditions and limits are surveyed and substantiated;
• relevant documents (operating limits and conditions, maintenance policy, symptom-based emergency operating procedures, other emergency procedures, emergency response plan) are surveyed and their modifications necessary to lifetime extension are justified.
• Upon the above activities it is ensured that during extended lifetime the safety function are fulfilled at the desired reliabilty, the safety analysis covers the possible operating modes and the operating limits and conditions are in harmony with lifetime extension requirements.

The followings shall be attached to the OL application: actualised FSAR, modified version of the above documents, the necessary special authority contributions. Background documentation to the substantiating documents shall be submitted upon further regulatory request.

Re-licensing of operating and other licenses expired at the end of lifetime

Conditions for the issuance of the operating license: the temporary storage or final disposal of radioactive wastes and spent fuel shall be ensured in harmony with the international expectations and experience. The valid operating license is precondition; maximal length is the operating license of the unit. In the application the followings shall be demonstrated:
• The operation is in accordance with the approved safety analysis.
• The inspection, manual and emergency documents and procedures are appropriate for safe

operation.
• Necessary initial data for condition monitoring of the SSCs are available.
• Safe operation is ensured fulfilling the operating limits and conditions.
• Technical and administrative conditions are ensured for long term safe operation, the financial resources performing long term maintenance and development of safety are available, the possible reasons for cancellation of the license are eliminated.
• The documents and contributions needed to OL are also parts of this application.

Guidelines relevant to lifetime extension
Besides the legally binding requirements the regulator has the possibility to issue not legally binding requirements. However these regulatory guidelines has important role in the system of regulations, because if the Licensee would like to deviate from the given guideline than it shall be justified that the applied method is more or at least equally conservative than the one of the guideline. This method shall be well substantiated.

The system of guidelines follows the structure of the Nuclear Safety Regulations; that is all of them are attached to one of the volume of the NSRs. So for example concerning ageing there are 4 guidelines in which four different aspects of requirements are address. In the guidelines the requirements of the NSRs are explained in details or the method of meeting the given requirement is formulated.

Concerning lifetime extension the following guidelines were already issued:

Maintenance
1.19 Inspection of the efficiency of the maintenance program of the nuclear power plant
4.7 Nuclear power plant maintenance program and maintenance efficiency monitoring
Ageing
1.26 Regulatory Inspection of the Ageing Management Program
2.15 Quality Assurance in the Ageing Management of Nuclear Power Plant Equipment
3.13 Consideration of Ageing during Nuclear Power Plant Design
4.12 Management of Ageing During Operation of Nuclear Power Plants
Equipment qualification
1.27 Regulatory control over equipment qualification and preservation of the qualified status
3.15 Equipment qualification requirements during the design of nuclear power plants
4.13 Equipment qualification requirement for operating nuclear power plants

Additionally the two most relevant guidelines are under issuance. These guidelines directly address the lifetime extension. The titles and numbers will be:
1.28 Requirements for the scope of the lifetime extension licence application
4.16 Conditions of operation licence renewal of nuclear installations

3. Every ten years the Licensee shall submit a Periodic Safety Review report, which is required for license retention.

Seq. No	Country	Article	Ref. in National Report
30	Canada	Article 8	2.2.2, page 24
Question/ Comment	The report indicates that “the Authority together with Paks NPP itself introduced the system of safety indicators”. Please provide detailed information on the safety indicators, and the criteria used by the HAEA staff to assess the performance of Paks NPP.		

Answer HAEA and Paks NPP developed their safety performance indicator system based on IAEA TECDOC-1141. The Authority uses on 3 main fields 9 overall indicators, 22 strategic indicators and 57 (quantitative) specific indicators.

The main fields are:

• Plants Operate Smoothly (6 strategic PIs based on 17 specific indicator)
• Plants Operate with Low Risk (7 strategic PIs, 18 specific indicator)
• Safety Attitude (9 strategic PIs, 22 specific indicator)

Clear and simple definition was created and thresholds were defined for each specific indicator. The value and the trend of indicators are assessed yearly by color coding.
The performance can be expressed by numbers– as the result of the evaluation applying engineering tools and safety judgment – in many cases only in comparison with the similar performance indicators of the previous years.
The structure of the main areas to be evaluated, the overall indicators, the strategic indicators and specific indicators can be seen on the figure No. 1-3.
The evaluation made is based on the criteria written in the "Safety Performance Evaluation Handbook" of the NSD. There is an example for the threshold and assessment of indicators.

Main Field: Safety Attitude
Overall indicator Compliance with instruction
Strategic indicator: Violation of instruction
Specific indicator: Number of Technical Specification exemptions

Definition: Number of temporary exemptions from the requirements of the OL&C licensed by the authority, apart from that the subject of the current case is to be reviewed in the future.

Assessment of indicator results:
Green: < 5 exemptions/year
Yellow: 5-25 exemptions/year
Red: >25 exemptions/year

Three color-marked evaluator fields were determined for the safety contributors as follows:

• „Green": The green field of the safety contributor spreads to the limit deemed appropriate by the authority. The values of the green field considered as reassuring by the authority and further action or increased attention is not deemed to be necessary.
• „Yellow": The limits of the warning, yellow field call the attention to the deviation from the desirable values within the regulatory acceptable range. The contributors within the yellow field should be paid increased attention and further actions should be done to avoid the prospective violation of the regulatory limits.
• „Red": The lower limit of the red, not acceptable field of the safety contributor is the limit value approved by the authority in different kind of documents - if such value is available – or in the event of lack of this value such individually determined criterion, in case of in-compliance with that the authority expects the corrective measure of the Licensee or itself initiates corrective measure.

The strategic indicators hold together the relevant, but not specific indicators, therefore the color evaluation of each safety indicator is based on the „worst" color of the specific indicators involved by itself.

The main fields and overall indicators are not characterized by colors, since in accordance with the NSD's opinion they require much more general assessment, which besides the quantified safety contributors is also influenced by information that are gathered from other sources and cannot be characterized by either quantities or indicators.

I. Plant operates smoothly

1.1. Operational performance

1.1.1. Unplanned shut downs and power reductions

- 1.1.1.1. Power reductions due to internal causes
- 1.1.1.2. Plant capacity factor

1.1.2. Maintenance planning

- 1.1.2.1. Ratio of planned and real duration of main outage
- 1.1.2.2. Ratio of work instructions beyond plan

1.2.State of systems and components

1.2.1. Maintenance

- 1.2.1.1. Maintenance on components classified to SCS
- 1.2.1.2. Ratio of preventive and total maintenance
- 1.2.1.3. Ratio of unsuccessful safety reviews

1.2.2. Material condition

- 1.2.2.1. Use-up of stressor cycles
- 1.2.2.2. Ratio of plugged SG tubes
- 1.2.2.3. Foreign materials

1.2.3. State of physical barriers

- 1.2.3.1. Fuel reliability
- 1.2.3.2. Primary leakage
- 1.2.3.3. Containment leakage

1.3. Events

1.3.1.Reportable events

- 1.3.1.1. Immediately reportable events
- 1.3.1.2. Reportable events
- 1.3.1.3. Indirectly reportable events
- 1.3.1.4. Event investigations ordered by NSD

II. Plant operates with low risk

2.1. Safety systems and components

- **2.1.1. Actual operation of safety systems**
  - 2.1.1.1. SCRAMs at nominal power
  - 2.1.1.2. Total number of SCRAMs
  - 2.1.1.3. SCRAM-III actuation
  - 2.1.1.4. ECCS operations
- **2.1.2. Availability**
  - 2.1.2.1. Unavailability detected during tests
  - 2.1.2.2. Diesels availability
  - 2.1.2.3. Pumps availability
  - 2.1.2.4. Availability of safety systems

2.2. Preparedness

- **2.2.1. Operational preparedness**
  - 2.2.1.1. Time devoted to training
  - 2.2.1.2. Ratio of unsuccessful regulatory exams
- **2.2.2. Emergency preparedness**
  - 2.2.2.2. Rate of participants in ERO training
  - 2.2.2.2. Deficiencies found during regulatory inspections

2.3. Risk

- **2.3.1. Operational risk**
  - 2.3.1.1. Number of TecSpec violations
  - 2.3.1.2. Number of occurrences under the effect of TecSpec
- **2.3.2. Calculation risk**
  - 2.3.2.1. Core-melting index
- **2.3.3. Environmental risk**
  - 2.3.3.1. Airborne radioactive release
  - 2.3.3.2. Liquid radioactive release
  - 2.3.3.3. Solid radioactive waste generated

III. Plant operates with a positive safety attitude

3.1. Compliance with instructions

3.1.1. Departure from planned state
- 3.1.1.1. Exemptions from the scope of the TecSpecs
- 3.1.1.2. Temporary modifications
- 3.1.1.3. Operating instructions
- 3.1.1.3. TecSpecs modifications

3.1.2. Violations of instructions
- 3.1.2.1. Number of TecSpec violations
- 3.1.2.2. Tests cancelled
- 3.1.2.3 Violations of licensing conditions

3.1.3. Deviation in reporting system
- 3.1.3.1. Delay of notification in case of immediately reportable events
- 3.1.3.2. Delay of notification in case of reportable events
- 3.1.3.3. Delay of submitting of investigation reports (30 days)

3.2. Human performance

3.2.1. Efficiency of radiation protection program
- 3.2.1.1. Eventual reports connecting to radiation protection
- 3.2.1.2. Dispersion of contamination
- 3.2.1.3. Work programs at high radiation level
- 3.2.1.4. Collective dose

3.2.2. Efficiency of industrial safety program
- 3.2.2.1. Works injuries
- 3.2.2.2. Fires

3.2.3. Human factor
- 3.2.3.1. Unsuitable state for work
- 3.2.3.3. Incidents caused by human failure

3.3. Striving for improvement

3.3.1. Self assessment
- 3.3.1.1. Number of independent internal audits

3.3.2. Corrective measures
- 3.3.2.1. Corrective measures of investigations
- 3.3.2.2. Corrective measures of QA audits

3.3.3. Experience feedback
- 3.3.3.1. Recurrent events

Seq. No	Country	Article	Ref. in National Report
31	Croatia	Article 8	Annex 7, p.118

Question/ Comment

Could you provide more information what changes will be made in the HAEA NSD according to IAEA recommendations and critical self-assessment of the tasks, resources, organization and working procedures of the HAEA NSD?

Answer

Results of internal investigations
Several key decisions and measures have been highlighted and they are briefly summarised below.

Changes in the relationship between the Authority and the Licensee
Within the framework of an extraordinary management meeting of Paks NPP and NSD the head of NSD repeatedly declared that the utmost priority above all is nuclear safety. He called upon the management of Paks NPP to make a declaration regarding the utitlity priorities. He emphasised that in addition to the declaration he expects Paks NPP to continuously demonstrate their commitment to safety through their actions.
As a result of discussions between NSD and Paks NPP the Licensee brought measures which aim to:
- improve the quality of submittals,
- decrease the number of urgent cases, and
- decrease the number of submittals to the Authority.

Changes in operational processes and the flow of information
The head of NSD brought a decision to review the operational processes of licensing and supervision. Within this framework the following tasks were defined:
- introducing a vertical method for licensing and supervision,
- developing a thematic basis for allocating tasks,
- reviewing the process of granting licence,
- reviewing or developing relevant Rules of Procedure and Guides.

The development of operational processes is under way. We developed the definition of vertical responsibility. This means that one organisational unit handles technical problems from their occurrence until their resolution and also takes care of the licensing and supervisory tasks – in co-operation with other organisational units, if necessary.
In order to improve the flow of information between organisational units the head of NSD set up several forums:
The Morning Management Meeting has been functioning before: this is a daily operative meeting to provide for the exchange of information between organisational units in order to support daily work.
The Meeting of the Advisory Committee held weekly: this forum serves the purpose of preparing decisions, negotiation and consultation. In addition to the meetings it aims to set tasks, ask for reports, and enhance the flow of information and co-operation among organisational units.
The Technical Meeting held monthly: its aim is to inform the technical staff of NSD about the status of current activities and the related issues and concerns.

Implementing the recommendations by the IAEA expert mission

The management of NSD defined the tasks of NSD resulting from the recommendations of the IAEA expert investigation of the incident, the responsible persons and the deadlines for their implementation. Part of these tasks overlap with some of the recommendations of the previous IAEA IRRT investigation and of the EU's RAMG project; on the other hand with short-term and long-term tasks previously identified by NSD.
The IAEA plans a follow-up mission between February 21 – March 1, 2005.

Recruitment of employees
The investigations among other things highlighted the restricted human resources of the Authority. The management of HAEA managed to increase the headcount of NSD by 5 employees. Due to the restricted salary made available by the wage table of civil servants we were not able to settle an agreement with applicants having professional experience so we hired career starters. Their training has been started.

Introducing new operational processes to evaluate submittals
We introduced a trial process for assessing applications for license in principle on modifications. This is a written methodology to assess and evaluate submittals and compliance with requirements and to provide proper documentation of the findings. On the basis of experience a decision will be brought at the end of this year concerning the future use of this methodology together with possible amendments and its application in case of other types of licences.

Monitoring tasks
In order to improve the fulfilment of tasks before the due date we introduced a task monitoring system operating as part of the quality assurance system. With the help of this system the management can continuously monitor the phases of implementation.

Decision on strategy development
The investigations pinpointed several gaps in the targeted, planned and structured implementation of activities. Following the review of the planning activity of NSD a proposal was compiled concerning strategy development. On the basis of this proposal the management of NSD assigned the task of developing a concept for strategy building to the head of the Strategy Department.

Restructuring the organisation
In order to remedy the identified shortcomings and problems to the most possible extent, to make the regulatory work of HAEA more effective and in line with international recommendations, to optimise resources available for regulatory work, to strengthen the strategic planning activities of NSD and to develop standards for handling nuclear and radiological incidents the head of NSD made a proposal to restructure the organisation of NSD.
As a result of negotiations the NSD functions on the basis of a new organisational structure from 13 September 2004. The introduction will be assessed and the necessary amendments and actions will be defined in April 2005.

Seq. No 32	Country Romania	Article Article 8	Ref. in National Report
Question/ Comment	Regarding the financial resources of the HAEA, could you mention if there are discrepancies between the salaries of the regulatory body staff and those of the		

	utility staff.
Answer	According to a comparison made in early 2004 junior and subordinate staff members of the regulatory body and engineers in similar positions with the NPP have comparable income. Lower- and medium level managers at the NPP earn about 40% more than the section heads and senior advisers with HAEA, department heads at the NPP have more than 60% higher income than managers in similar positions with HAEA. There are no data available on the highest level management. The problem is characterised by the fact, that when in 2004 HAEA opened five new positions within the regulatory body, no experienced staff members could be recruited, all five positions have been taken by recently graduated engineers.

Seq. No	Country	Article	Ref. in National Report
33	France	Article 8.1	p.20 & 34

Question/ Comment	The second national Report of Hungary (2001) mentioned in its page 34 that "the wages of the Authority staff, in contrast with international demands, (were) significantly lower than the wages of the nuclear power plants". The third national report (neither p. 20 nor p. 34) come back on this issue. Could Hungary state upon the resolution of that issue or possibly remaining problems?
Answer	According to a comparison made in early 2004 junior and subordinate staff members of the regulatory body and engineers in similar positions with the NPP have comparable income. Lower- and medium level managers at the NPP earn about 40% more than the section heads and senior advisers with HAEA, department heads at the NPP have more than 60% higher income than managers in similar positions with HAEA. There are no data available on the highest level management. The problem is characterised by the fact, that when in 2004 HAEA opened five new positions within the regulatory body, no experienced staff members could be recruited, all five positions have been taken by recently graduated engineers.

Seq. No	Country	Article	Ref. in National Report
34	Japan	Article 8.1	P.26/L.1

Question/ Comment	It is reported in P.26 that press releases related to the occurrence of incidents are published by the operators and the Authority's participation is only in relation to the INES classification of events. The procedure on the event report from the licensees to regulatory body is reported in P.78. Could you explain details of information disclosure to public, especially the regulatory response to the corrective actions performed by the licensees?
Answer	Normally the Authority does not disclose the status of the corrective actions directly to the public, since most of such actions are not of interest. On the other hand events that arise the interest of the public are openly discussed with the media both on request and by press releases and this is one way how HAEA informs the public. Another way is offered by the web-site of HAEA where every news related to the activity of HAEA that may be of common interest are posted, while the major new agencies are informed on it. Besides HAEA yearly reports to the Parliament and to the government.

Seq. No	Country	Article	Ref. in National Report
35	United States of America	Article 8.1	Section 3.4.1

Question/ Comment	Section 3.4.1 discusses current staff and training procedures, but not plans for recruitment. What steps is HAEA taking to ensure appropriate staff and knowledge transfer given staff retirement (e.g., the licensee is recruiting young people

systematically)?

Answer	The number of the HAEA personnel is determined by its supervising ministry. Although the number of staff is usually quite stable, in 2004 HAEA had the opportunity to recruit 5 new employees and the average age of the staff is quite low. The new employees are trained based on a predefined two-year-long introductory training program. As a part of the training program the new inspectors also receive training from the senior experts of the HAEA, and there are special on-the-job trainings held by acknowledged experts. There is a process at the HAEA to measure the overall knowledge profile of the staff. All employees have to fill in a questioner within a Lotus Notes database, and make a statement about their fields of expertise. This survey covers all theme and knowledge in the area of nuclear safety. The database will give assistance to HAEA for the planning of the future training programs. According to what has been said above, the present staffing situation is acceptable, however, in the long run replacement of the retiring personnel and need for new staff for the coming new tasks shall be a concern. Legal as well as organisational steps to cope with this issue have been initiated by the management of HAEA.

Seq. No	Country	Article	Ref. in National Report
36	France	Article 8.2	p. 19

Question/ Comment	The reports mentions that "the Authority's scope of competence comprises nuclear safety licensing …. and co-ordination of research and development". Could Hungary specify what is exactly the field covered by this research and development? Does-it includes operating nuclear research facilities?
Answer	According to the Act on nuclear energy it is the HAEA's obligation to finance those research and development topics which are directly related to regulating and inspecting nuclear safety. Any research and development, however, which is related to enhance the efficiency and prove the safety of existing or planned nuclear facilities, is the responsibility of the industry. The existing nuclear research facilities in Hungary are mostly engaged in research programs outside the field of nuclear technologies (e. g. solid state physics, medical etc.) and only a few (e. g. the thermal-hydraulic stand) are used for nuclear technology research, but it is mostly financed by the industry. HAEA is regularly informed about the results of these latter research project, as well as of the research projects which are carried out in the framework of various international projects (e. g. IAEA, EU, OECD NEA). In Hungary, there are no research facilities which are related to developing new nuclear technologies.

Seq. No	Country	Article	Ref. in National Report
37	Bulgaria	Article 9	page 31

Question/ Comment	Were any changes introduced to the Safety policy requirements after the second report especially with regard to lessons learned from the Paks 2003 incident?
Answer	The Safety Policy document of the plant was revised and modified together with the entire system of policy documents of the company in the frame of the Program of Organisation Development. (i.e. Mission and Values, Company Strategy, Functional Strategies, Policies (including Safety Policy)). However it is not correct to say that the deficiencies in Safety Policy contributed to the Paks 2003 incident. Specific requirements were missing from lower level procedures or, were not followed if they existed.

Seq. No	Country	Article	Ref. in National Report

38	Germany	Article 9	p. 23, 2.2.2
Question/ Comment	The HAEA is certified according to ISO 9001. Is there any intention of the Paks NPP to perform a certification process, too?		
Answer	The requirements of ISO 9001:2000 and Paks NPP Quality System (2nd Volume of Nuclear Safety Regulations (NSR)) have been compared. The NPP determined the extra requirements specified by ISO which are progressive and seem to be reasonable to be taken into account by Paks NPP. The NPP Quality System is in accordance with the 2nd Volume of the NSR. As a result of the ongoing development of the management system the Quality System shall incorporate the extra requirements by the end of 2005. In general it can be stated that the system also fulfils the recommendations by the IAEA safety series document 50-C/SG-Q. It should also be mentioned that regarding the Environmental Management System the Paks NPP has the qualification ISO 14000.		
Seq. No 39	Country Germany	Article Article 9	Ref. in National Report p. 25, 2.2.3
Question/ Comment	Is it planned to follow the trend to a risk-informed and performance-based approach to regulation complementary to the compliance-based approach?		
Answer	Yes, it is planned to follow the trend to a risk-informed approach to regulation complementary to the compliance-based approach. For this purpose an Implementation Plan was prepared and approved at the HAEA and a comprehensive long term project was launched in 2003. The Risk-informed Implementation Project (RIP) schedules all the tasks, which have been identified important to improve the legislative, modelling and training areas as prerequisites for the successful implementation.		
Seq. No 40	Country Germany	Article Article 9	Ref. in National Report p. 25, 2.2.3
Question/ Comment	Is it planned to implement regulations or guidelines on safety management?		
Answer	A very important safety management tool is the quality management system, which keeps control over all activities, especially those, which are safety related. The HAEA NSD has applied the IAEA 50-C-QA guide, and in the Nuclear Safety Regulation (NSR) many ISO-9000-2000 standard statements have been incorporated. The new series of Safety Regulations (the review of which has been completed in 2004) put more emphasis on the responsibility of the NPP (or other nuclear installation) management. NSR Volume 2. (on Quality Management) 2.012. prescribes: Nuclear safety is the fundamental aspect at identification of all such processes, services and products which are covered by the quality management system. NSR Volume 4. (on Operation) with the authorisation by the Atomic Energy Act, among other requirements, prescribes: 5.002. The organization shall be established in accordance with the acts, legal regulations and rules in force. Tasks to be performed by the organization shall be identified. Activities needed for execution of tasks shall be regulated in appropriate		

documents. The hierarchy of responsibilities, authorization needed for execution of tasks, complement of the staff and requirements for their skills shall be specified in these documents. The organizational units shall be staffed to the necessary extent with personnel suitable for execution of their tasks at adequate level.

5.003. At establishment of organizational units, functions of the organization shall be considered and it shall be ensured that selection and assignment of the staff (including the staff employed under a contract) is in accordance with fulfilment of functions. Persons assigned to lead these organizational units are simultaneously responsible for safety aspects of the activity performed by the organizational unit. The most important aspect of establishment of the organization is that it has to provide the safe operation of the plant in every operating condition and possibly in an emergency situation.

5.004. The top manager of the plant is solely responsible for safety of the nuclear power plant. Selection of the top manager shall be performed on the basis of careful deliberation of professional, human, managerial and ethical features.

Seq. No	Country	Article	Ref. in National Report
41	Austria	Article 10	
Question/ Comment	Which cost factor criteria has the Authority to take into account ?		
Answer	The Nuclear Safety Authority does not have any cost related criterion when taking its decisions. The only criterion to be respected is nuclear safety. Financial aspects may influence a decision in the sense described in the answer to question no. 13 by Austria.		

Seq. No	Country	Article	Ref. in National Report
42	Austria	Article 10	
Question/ Comment	How does the responsibility to reduce risk while taking costs into account conflict with the principle of "priority of safety over all other aspects", and how does HAEA manage the distinction between considering costs and giving safety priority over all other matters in making regulatory decisions concerning plant modifications?		
Answer	In the National Report paragraph 3.1.3 writes: It is the responsibility of the Licensee to keep risks down to an appropriate level. In the field of safety improvement measures, however, the Authority should also set a priority list. Priorities should be examined not only from the viewpoint of the reduction of risk but also taking costs into account. The above statement is referring to one of the paragraphs of the HAEA’s Safety Policy and Basic Principles of Operation: The permanent reduction of risk is the task of Licensee. However, in the field of the safety upgrading, HAEA NSD should also have a priority list. Priority has to be examined in relation of risk reduction vs. cost. The “risk reduction versus cost” principle is in accordance with the ALARA or ALARP principles, and remains conform to the environment protection and industrial safety standards and safety principles. The cost is mentioned in the context of prioritisation of safety upgrading measures. That means, the Regulator should have a view on the risk reduction possibilities, and an idea about the resources they need. The perception of the resource or cost required is a kind of rough engineering judgement.		

Since the zero risk implies infinite costs, and the unconditional priority of safety over all other aspects (including the consumption needs of the society) means immediate closure and decommissioning of all nuclear and non-nuclear but any industrial installations, one has to apply realistic principles.

Keeping in mind the societal needs the benefits and products of the different industries, there must be a priority list of safety issues, which list at the side of the Regulator not necessarily equal to that priority list, what the Licensee has. The prioritisation may consider cost aspects, considering not only the risk, but the cost tolerance of the wide public (inspection and product costs versus the safety of the demanded production).

Since the expectation against the nuclear energy is that, it should not impose bigger risk to the public, then other industries, and the zero risk is not among the realistic targets, the above principles could be applied to nuclear installations, too.

This kind of cost-benefit view is not applied by the regulator during the licensing of technically outlined projects, or giving permissions for system modifications or equipment changes. The licensing documentations do not have cost calculation chapters, while the safety inspectors are not really prepared for cost analysis. Due to these obstacles there are not cost estimations considered in the nuclear safety regulatory decisions and resolutions.

Seq. No	Country	Article	Ref. in National Report
43	Finland	Article 10	chapter 3.2

Question/ Comment: Safety policy and organizational arrangements for safety are described in paragraph 3.2. Safety Committee is not mentioned in 3.2 and Annex 7. What is the role of Safety Committee at the PAKS NPP?

Answer: The company has a committee called Safety and QA Management Committee. The main safety related topics to be discussed and evaluated by this committee are the following:

- Managing deviations from the existing basic documents (FSAR, PSR reports etc)
- Review of the status of modifications of the Technical Specifications
- Discussion of any questions or suggestions related to the nuclear safety of the plant.
- Review of the Safety Indicators' trends and proposing corrective actions.
- Review of the in-house and industry operating experience.

Seq. No	Country	Article	Ref. in National Report
44	Finland	Article 10	

Question/ Comment: The investigations of Paks fuel damage accident have indicated that production and short term economical objectives have been prioritized over safety in a number of decision making situations. Signs of gradual deterioration of the safety culture and physical condition of the plant have been observed over a time period of more than ten years. It seems that the Paks top management has not adequately met its safety responsibility. The negative trend is evidently caused by frequent changes of Paks management, and political appointment of top managers who lack previous experience from nuclear power operation.

	Could you explain what are the current practices and qualification requirements in appointing the responsible manager of Paks NPP? Is the regulatory body involved in assessing whether a candidate for top management position of Paks has the capability to meet the responsibility for safe nuclear operation?
Answer	The Director General of the Paks NPP is appointed by the Board of Directors. The members of the Board are elected by the General Assembly of the shareholder company. More than 99% of the shares are with the state owned Hungarian Electricity Works (HEW). The appointment of the top manager of the plant is sometimes influenced also by political considerations. In a recent change at the top manager position of the Paks NPP one of the technical directors (formerly having gone through all the grades of technological plant-positions) has been appointed to Director General. The nuclear regulatory body has no legal means to influence the decision on the selection of plant managers and so far was not involved in any (unofficial) way either. The head of HAEA NSD has recently initiated an exchange with the head of the Board of Directors of the HEW on the possible involvement of the regulatory body in the appointment procedure with no answer so far.

Seq. No	Country	Article	Ref. in National Report
45	France	Article 10	§3.1.3 - p. 30

Question/ Comment	Could Hungary indicate which criteria are used by the Authority to set up priorities from the viewpoint of risk reduction vs costs?
Answer	In the National Report paragraph 3.1.3 writes: It is the responsibility of the Licensee to keep risks down to an appropriate level. In the field of safety improvement measures, however, the Authority should also set a priority list. Priorities should be examined not only from the viewpoint of the reduction of risk but also taking costs into account. The above statement is referring to one of the paragraphs of the HAEA's Safety Policy and Basic Principles of Operation: The permanent reduction of risk is the task of Licensee. However, in the field of the safety upgrading, HAEA NSD should also have a priority list. Priority has to be examined in relation of risk reduction vs. cost. The "risk reduction versus cost" principle is in accordance with the ALARA or ALARP principles, and remains conform to the environment protection and industrial safety standards and safety principles. The cost is mentioned in the context of prioritisation of safety upgrading measures. That means, the Regulator should have a view on the risk reduction possibilities, and an idea about the resources they need. The perception of the resource or cost required is a kind of rough engineering judgement. Since the zero risk implies infinite costs, and the unconditional priority of safety over all other aspects (including the consumption needs of the society) means immediate closure and decommissioning of all nuclear and non-nuclear but any industrial installations, one has to apply realistic principles. Keeping in mind the societal needs the benefits and products of the different industries, there must be a priority list of safety issues, which list at the side of the

Regulator not necessarily equal to that priority list, what the Licensee has. The prioritisation may consider cost aspects, considering not only the risk, but the cost tolerance of the wide public (inspection and product costs versus the safety of the demanded production).

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Seq. No 46	Country Germany	Article Article 10	Ref. in National Report p. 31, 20 and 41
Question/ Comment	This question also relates to Articles 8 and 13 (chapters 2.2 and 3.6.3) Which measures have been implemented in the HAEA decision-making process with respect to the lessons learnt from the Paks-2 accident April 2003 in order to avoid repetition?		

Answer

The management of NSD reviewed its safety policy and the document governing its operating rules with a view on the recommendations by the IAEA expert mission invited to Hungary following the incident. A decision making policy and procedure are under elaboration. In addition HAEA NSD has summarised the results of the internal investigations and highlighted several key decisions and measures that are briefly summarised below.

Changes in the relationship between the Authority and the Licensee
Within the framework of an extraordinary management meeting of Paks NPP and NSD the head of NSD repeatedly declared that the utmost priority above all is nuclear safety. He called upon the management of Paks NPP to make a declaration regarding the utitlity priorities. He emphasised that in addition to the declaration he expects Paks NPP to continuously demonstrate their commitment to safety through their actions.
As a result of discussions between NSD and Paks NPP the Licensee brought measures which aim to:
- improve the quality of submittals,
- decrease the number of urgent cases, and
- decrease the number of submittals to the Authority.

Changes in operational processes and the flow of information
The head of NSD brought a decision to review the operational processes of licensing and supervision. Within this framework the following tasks were defined:
- introducing a vertical method for licensing and supervision,
- developing a thematic basis for allocating tasks,
- reviewing the process of granting licence,
- reviewing or developing relevant Rules of Procedure and Guides.
The development of operational processes is under way. We developed the

definition of vertical responsibility. This means that one organisational unit handles technical problems from their occurrence until their resolution and also takes care of the licensing and supervisory tasks – in co-operation with other organisational units, if necessary.
In order to improve the flow of information between organisational units the head of NSD set up several forums:
The Morning Management Meeting has been functioning before: this is a daily operative meeting to provide for the exchange of information between organisational units in order to support daily work.
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The IAEA plans a follow-up mission between February 21 – March 1, 2005.

Recruitment of employees
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Introducing new operational processes to evaluate submittals
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Monitoring tasks
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Decision on strategy development
The investigations pinpointed several gaps in the targeted, planned and structured implementation of activities. Following the review of the planning activity of NSD a proposal was compiled concerning strategy development. On the basis of this

	proposal the management of NSD assigned the task of developing a concept for strategy building to the head of the Strategy Department. Restructuring the organisation In order to remedy the identified shortcomings and problems to the most possible extent, to make the regulatory work of HAEA more effective and in line with international recommendations, to optimise resources available for regulatory work, to strengthen the strategic planning activities of NSD and to develop standards for handling nuclear and radiological incidents the head of NSD made a proposal to restructure the organisation of NSD. As a result of negotiations the NSD functions on the basis of a new organisational structure from 13 September 2004. The introduction will be assessed and the necessary amendments and actions will be defined in April 2005.		
Seq. No 47	Country Germany	Article Article 10	Ref. in National Report p. 31, 3.2
Question/ Comment	Is there a data acquisition and processing system to evaluate whether the safety improvement measures have been beneficial to overall safety?		
Answer	The plant has a living PSA model and a reliability database. The reliability data of the components is systematically collected and the input data for PSA analysis is regularly calculated and updated. The existing PSA studies are updated annually. All the safety related plant modifications and changes in the reliability characteristics of plant equipment and/or plant personnel are modelled, the PSA results and documentation are updated as necessary. According to these assessments, the reported considerable risk reduction can be principally attributed to the safety enhancement measures that have been implemented at Paks NPP up to now. In addition, there were also changes in the understanding of the plant, due to operational experience, data collection, and certain improvements in the modelling techniques. According to this, the risk measures also have changed but in a much less significant way and in certain cases the calculated results even have increased. Some of the improvements that were effective in risk reduction are listed here: • Relocation of emergency feed water system, at the recent location the system is now protected from hazard of high-energy line breaks and of fires and floods in the turbine hall; • Protection of containment sump against clogging with redesigning of the sump strainers; • Prevention of the refilling of the tanks of the low-pressure emergency core cooling system after they have been emptied; • Elimination of the so-called artificial voltage cutting; • Modification of the primary pressure relief system, introduction of “bleed and feed” possibility, realisation of a protection against cold overpressure; • Modification of the reactor protection system with introduction of new protection functions and operating conditions, applying consistently specific design principles.		
Seq. No 48	Country Germany	Article Article 10	Ref. in National Report p. 31, 3.2 and 3.3.2
Question/ Comment	This question also relates to Article 11.1 Is there a system for tracking possible safety impacts caused by deregulations of the energy market?		

Answer The Hungarian regulator has already established a wide set of safety performance indicators, in accordance with the international good practice. However it is very hard to distinguish, whether in case of declining parameters the reason is poor management performance or market effects when short term financial interests are influencing the management decisions.

The potential energy market effects are not necessarily specific to Hungary, however we can list the characteristic types according to our experience.

The unit configuration requirements and the system non availability limitations in the Technical Specification (TS) may have conflicts with the readiness to follow the orders of the grid operator. This readiness has become more important than several years ago through the tariff constructions, which reward peak generation, hot stand by, compensate when the generation is lower comparing to the nominal level written in the commercial contract, and penalise when the generation doesn't meet the demand.

No systematic tracking of deregulation effects has yet been installed, since no actual safety impact of the deregulation has so far been experienced.

The importance of the scheduled electric power generation has significantly increased according to commercial contracts. Both the constant electric power generation demand with the highest capacity and the load dropping of one or more nuclear units due to grid control aspects are present during the daily manoeuvres. The readiness to follow the orders of the grid operator has become more important than several years ago through the tariff constructions, which reward peak generation, hot stand by, compensate when the generation is lower comparing to the nominal level written in the commercial contract, and penalise when the generation doesn't meet the demand.
Since the necessary unit configuration prescribed in the Technical Specification Operational Limits and Conditions, (TS OLC) the non availability of the systems and equipment due to random errors is controlled through time limitations. The prescription is that, after shorter or longer time, which time parameter is system specific, if the operator could not succeed with the repair or restore activity, the unit should be stopped with decreased reactor power to certain levels, depending on the available unit configuration. There are important requirements on periodic tests and maintenance activities, too.
The operator may try to keep the unit on-line with nominal power, while it can not succeed with repair of random error during that time span which is prescribed in the TS OLC on non availability of the specific system, degraded due to the random error.
Alternatively the operator may intended to postpone periodic tests or periodic maintenance activity, because they may be in conflict with the generating demand. Such situations are well known in the international practice. The situation controlling tools of the Hungarian regulator are as follows:
• Enforcement action, which may lead to penalty
• Harnessing of the publicity and moral effect of the safety performance periodic and yearly evaluation
• With systematic R+D the availability parameters, periodic test and periodic maintenance requirements of the OLC could be refined. In many cases the originally

determined availability, test and maintenance requirements are too conservative, or they lack detailed substantiation, just set on the basis of technical traditions.
Other market effect is the presence of contracted companies in the NPP. The problems related to this issue are again not specific to Hungarian, but common in the international environment. The regulatory reaction could be characterised with the surveillance, enforcement action if it is necessary, which may lead to penalty, and stressing the primary responsibility of the NPP management and the native personnel, even in the regulations.

Seq. No	Country	Article	Ref. in National Report
49	Germany	Article 10	p. 31, 3.2

Question/ Comment

This question also relates to Article 12 (p. 38, 3.5.5 and Article 14, 3.7.3)
Is there a performance indicator system or a similar system for safety management and safety culture, especially for possible identification of degradation, at an early stage?

Answer

Development of current Safety Performance Indicators system of Paks NPP was completed in 2001 and was introduced in 1 January 2002. This system replaced the old set of Safety Performance Indicators in order to reflect the plant safety performance on the basis of a wider range and specially arranged system of indicators. The new system was developed on the basis of IAEA TECDOC-1141. Hierarchical structure of SPI system contains 4 levels (72 specific indicators, 20 strategic indicators, 8 overall indicators, 3 attributes). On the top of the structure there are three main safety attributes characterizing the operational safety performance of the plant. Indicators belonging to the attribute called 'Attitude towards safety' contains indicators which to some extent can characterize safety culture. For every indicator target values and thresholds for unacceptability were determined that help to assess trends. Results of SPI assessment are presented in the Quarterly Report of SPI in the meeting of Safety and QA Management Committee. A regular managerial assessment of the values of safety indicator system has been introduced. The development of a web based computer program to support the assessment work is in progress.

Seq. No	Country	Article	Ref. in National Report
50	Romania	Article 10	

Question/ Comment

Could HAEA provide information on the performance indicators and how are these used by the regulatory authority in assessing the plant safety.

Answer

HAEA developed their safety performance indicator system based on IAEA TECDOC-1141. The Authority uses 22 strategic indicators 57 (quantitative) specific indicators on 3 main fields. Thresholds were defined for each specific indicator. The value and the trend of indicators are assessed yearly by color coding.
The performance can be expressed by numbers– as the result of the evaluation applying engineering tools and safety judgment – in many cases only in comparison with the similar performance indicators of the previous years.

NSD evaluates the safety performance contributors against particularly determined criteria and fits them in evaluation color codes in accordance with the following:
• „green": The green field of the safety contributor spreads to the limit deemed appropriate by the authority. The values of the green field are considered as reassuring by the authority, the appropriate reserve is ensured in comparison with the regulatory requirements or with the possible declared regulatory limits, and further action or increased attention is not deemed to be necessary.

• „yellow": The limits of the warning, yellow field call the attention to the deviation from the desirable values within the regulatory acceptable range. In order to prevent the prospective violation of the regulatory limits the authority pays increased attention to the contributors falling into the yellow field.
• „red": The safety contributor within the red field is not acceptable. The lower limit of the red field is the limit value approved by the authority in different kind of documents – if such value is available – or in the event of lack of this value such individually determined criterion, in case of in-compliance with which the authority expects the corrective measure of the licensee or itself initiates corrective measure.
The result of the safety performance indicator is one of the basic contributor to the planning of annual inspection schedule.

Seq. No	Country	Article	Ref. in National Report
51	Romania	Article 10	

Question/ Comment	Are there any observed effects of the electricity market de-regulation on nuclear safety and what is the regulatory approach to this issue?
Answer	The importance of the scheduled electric power generation has significantly increased according to commercial contracts. Both the constant electric power generation demand with the highest capacity and the load dropping of one or more nuclear units due to grid control aspects are present during the daily manoeuvres. The readiness to follow the orders of the grid operator has become more important than several years ago through the tariff constructions, which reward peak generation, hot stand by, compensate when the generation is lower comparing to the nominal level written in the commercial contract, and penalise when the generation doesn't meet the demand. Since the necessary unit configuration prescribed in the Technical Specification Operational Limits and Conditions, (TS OLC) the non availability of the systems and equipment due to random errors is controlled through time limitations. The prescription is that, after shorter or longer time, which time parameter is system specific, if the operator could not succeed with the repair or restore activity, the unit should be stopped with decreased reactor power to certain levels, depending on the available unit configuration. There are important requirements on periodic tests and maintenance activities, too. The operator may try to keep the unit on-line with nominal power, while it can not succeed with repair of random error during that time span which is prescribed in the TS OLC on non availability of the specific system, degraded due to the random error. Alternatively the operator may intended to postpone periodic tests or periodic maintenance activity, because they may be in conflict with the generating demand. Such situations are well known in the international practice. The situation controlling tools of the Hungarian regulator are as follows: • Enforcement action, which may lead to penalty • Harnessing of the publicity and moral effect of the safety performance periodic and yearly evaluation • With systematic R+D the availability parameters, periodic test and periodic maintenance requirements of the OLC could be refined. In many cases the originally determined availability, test and maintenance requirements are too conservative, or they lack detailed substantiation, just set on the basis of technical traditions. Other market effect is the presence of contracted companies in the NPP. The problems related to this issue are again not specific to Hungarian, but common in the

international environment. The regulatory reaction could be characterised with the surveillance, enforcement action if it is necessary, which may lead to penalty, and stressing the primary responsibility of the NPP management and the native personnel, even in the regulations.

The Hungarian regulator has already established a wide set of safety performance indicators, in accordance with the international good practice. However it is very hard to distinguish, whether in case of declining parameters the reason is poor management performance or market effects when short term financial interests are influencing the management decisions.

The potential energy market effects are not necessarily specific to Hungary, however we can list the characteristic types according to our experience.

The unit configuration requirements and the system non availability limitations in the Technical Specification (TS) may have conflicts with the readiness to follow the orders of the grid operator. This readiness has become more important than several years ago through the tariff constructions, which reward peak generation, hot stand by, compensate when the generation is lower comparing to the nominal level written in the commercial contract, and penalise when the generation doesn't meet the demand.

No systematic tracking of deregulation effects has yet been installed, since no actual safety impact of the deregulation has so far been experienced.

Seq. No 52	Country United States of America	Article Article 10	Ref. in National Report Section 3.1
Question/ Comment	Section 3.1 states that Aseveral assessments were performed by the Authority to survey the operator=s safety culture.@ Were these assessments performed before or after the April 2003 fuel handling incident? What metrics were used to assess safety culture? What insights were gained from these assessments and the self-assessment of the Authority in late 2003, and how can they be applied to the incident described in Annex 7?		

Answer

The assessment activities are continuously performed by the authority, hence before and after the incident too. Safety indicators and the lessons learned from the incident analyses are used to monitor the safety culture of the operator.

The Safety attitude is one of the main areas of the Safety indicator system related to safety culture.
A Safety attitude area monitors the following three fields as overall indicators:
• compliance with instructions,
• human performance,
• striving for improvement

List of specific indicators on safety culture related area:
• Exemptions from the scope of the TecSpecs
• Temporary modifications
• Operating instructions
• Technical Specification modifications

• Technical Specification violations
• Delay of notification, report, and investigation report of events
• Violations of licensing conditions
• Eventual reports in connection with radiation protection
• Dispersion of contamination
• Number of work programs in high radiation
• Collective dose
• Works injuries
• Fires
• Unsuitable state for work / Individual non-fitness for duty
• Incidents caused by human error
• Corrective measures of investigations
• Recurrent events

Clear and simple definition was created and thresholds were defined for each specific indicator. The value and the trend of indicators are assessed yearly by color coding.
The performance can be expressed by numbers– as the result of the evaluation applying engineering tools and safety judgment – in many cases only in comparison with the similar performance indicators of the previous years.
The structure of the main areas to be evaluated, the overall indicators, the strategic indicators and specific indicators can be seen on the figure No. 1-3.
The evaluation made is based on the criteria written in the "Safety Performance Evaluation Handbook" of the NSD.

Three color-marked evaluator fields were determined for the safety contributors as follows:
• „Green": The green field of the safety contributor spreads to the limit deemed appropriate by the authority. The values of the green field considered as reassuring by the authority and further action or increased attention is not deemed to be necessary.

• „Yellow": The limits of the warning, yellow field call the attention to the deviation from the desirable values within the regulatory acceptable range. The contributors within the yellow field should be paid increased attention and further actions should be done to avoid the prospective violation of the regulatory limits.

• „Red": The lower limit of the red, not acceptable field of the safety contributor is the limit value approved by the authority in different kind of documents - if such value is available – or in the event of lack of this value such individually determined criterion, in case of in-compliance with that the authority expects the corrective measure of the Licensee or itself initiates corrective measure.

Before the fuel handling incident the Authority detected many times symptoms of degrading safety culture and noted them to the operator and prescribed improvement measures or the elaboration of them to the operator in each case.

During the post incident regulatory investigation and self-investigation the problems and weaknesses were identified and improvement measures were established whose implementation in majority was done or near to be done.

The operator elaborated an organization development program based on the results of the incident investigation, the regulatory and self-assessments. The main elements of the program are the review and enhancement of the procedures and documents, etc. The Authority approved the program and inspects regularly its execution.

The evaluation of the safety performance indicators for the year 2003 showed bad results. According to preliminary evaluation for the year 2004 the signs of improvement can be envisaged.

Seq. No 53	Country United States of America	Article Article 10	Ref. in National Report
Question/ Comment	How are allegations of safety concerns from the public or nuclear power plant workers handled by the regulatory body?		
Answer	The citizens including nuclear power plant workers may directly contact the regulatory body with questions pertaining to nuclear safety. The regulatory body shell answer any question within 15 days (in accordance with Act VXIII of 1992). Besides, the regulatory body regularly reports to the public and to its representatives (parliament, mass media etc.) in order to make information on nuclear safety matters available and transparent. The nuclear power plant workers are questioned also during regulatory inspections and their answers are taken into account in various regulatory decisions.		

Seq. No 54	Country Austria	Article Article 11	Ref. in National Report
Question/ Comment	The report does not provide details on the decommissioning of Paks NPPs. a) What is the estimated cost of decommissioning? b) Does the estimated cost of decommissioning include the costs of radioactive waste disposal and management of spent nuclear fuel until a repository is opened? c) At what rate is the decommissioning fund accumulating financial resources to pay for this cost? d) Is the rate of accumulation sufficient to ensure that the necessary amount of funding will be in place by the time the operating licenses expire? If not, what steps are being taken now to ensure that alternative sources of funding for decommissioning will be available when decommissioning activities begin? e) If the licensee defaults on decommissioning due to inadequate financial resources, upon whom does responsibility fall to undertake to complete decommissioning? f) Does the HAEA have a legal responsibility to ensure that adequate decommissioning funding exists, and if it determines that there is a shortfall, what are HAEA's legal remedies to seek redress of the situation?		
Answer	According to the basic rules, laid down in Act on Atomic Energy, radioactive waste management shall not impose undue burden on future generations. Accordingly, by the Act and its executive orders, a Central Nuclear Financial Fund was established on 1st of January 1998 to finance radioactive waste disposal, interim storage and disposal of spent fuel as well as decommissioning of nuclear facilities. The Fund is managed by the Hungarian Atomic Energy Authority (HAEA), the Minister supervising the HAEA is disposing over the Fund. The Paks NPP is the major contributor to the Fund. The long term plan – approved by the Minister supervising the HAEA – on the activities financed from the Fund (up to the decommissioning of the nuclear facilities and the disposal of the radioactive waste produced by the		

decommissioning) is annually reviewed and revised as required. The total cost of radioactive waste and spent fuel management as well as decommissioning is – discounted for 2004 – 296 000 M HUF (about 1180 M €), according to the calculations made in 2004. The payments of Paks NPP into the Fund are defined so, that during the lifetime of the Paks NPP (till 2017) the total amount of money should be collected. An eventual extension of this lifetime is not taken into consideration. The payments of Paks NPP into the Fund are prescribed year by year in Act on the Central Budget. The responsibility for the adequate funding rests therefore with the Parliament, not with the HAEA.
In order to ensure that the Fund maintains its value, the Government has to contribute to the Fund with a sum, calculated on the average assets of the Fund in the previous year using the average base interest rate of the central bank in the previous year.

Seq. No 55	Country Austria	Article Article 11	Ref. in National Report
Question/ Comment	How is it ensured that the Authority staff which is trained in the NPP is independent?		
Answer	The new employees at the Authority have a special, two-year-long training program. Besides the training at the power plant and other licensees they receive education from the HAEA's and other Hungarian experts. It is also possible for the new inspectors to participate in international training courses, in order to broadening their overview and knowledge. At the end of the training program the new inspectors have to take a final exam in front of a board of examination, which includes senior experts with great experience. After the exam the new inspectors may work individually. There are also annual training plans for the whole staff to ensure the maintenance of the acquired knowledge. Independence of the staff is ensured by the fact that besides taking part in training at the plant no relationship whatsoever may evolve between the trainees and the plant. Training of inspectors at the plant is indispensable and has a long tradition with HAEA NSD. During the long history of such trainings no sign of any kind has ever been experienced of partiality of any inspector.		

Seq. No 56	Country Austria	Article Article 11	Ref. in National Report
Question/ Comment	What steps have been taken, and what plans for the future exist, to ensure that sufficient "nuclear knowledge" is preserved to ensure safe conduct of plant operations (through and including decommissioning and final disposal of radioactive waste and spent fuel) and safe regulation of nuclear activities?		
Answer	In the case of the plant operation it is the task of the licensee to ensure existence of adequate "nuclear knowledge". As it is described in part 3.5.4. of our National Report implementation of this requirement is based on a strategic level thinking and on suitable organisational, procedural and tutorial bases supplemented with the (partly also referred) training infrastructure. It is worth to mention, that the chairman of the Board of the Paks NPP Ltd. for the moment is a professor of the Budapest University of Technology and Economics (BUTE) who has substantial activities in the university education of future engineers, and is fully aware of the long term needs in specialists of the all energy sector of the Hungarian economy. What concerns the human resources for legislative and regulatory activities, it is not		

based on a very long term comprehensive strategic plan, but instead on two main components. One is the cautiousness and mid-term foresight in recruitment and training of the regulatory body personnel. This may be characterised by employment of 5 young engineers last year in the HAEA NSD, who are now on practical shift trainings in the NPP (see part 3.4.1. of the National Report). The other element is the existence of academic and research centres serving as Technical Support Organisations for the regulatory body. (Among them the Institute of Nuclear Techniques within the BUTE has a specific importance with its training nuclear reactor, serving more than 30 years for education of generations of nuclear engineers.) This approach is based on the Act on Atomic energy which stipulates, that the technical activities serving as basis for the regulatory control shall be funded from the central budget.
(Note: questions of decommissioning [with the exception of its licensing and the regulatory authority for that] as well as final disposal of radioactive waste and spent fuel are beyond the scope of application of the Convention on Nuclear Safety.)

Seq. No 57	Country Czech Republic	Article Article 11	Ref. in National Report
Question/ Comment	Can you describe payment system to the Authority from licensees of nuclear installations? Which part of the Authority budget is covered by this payment?		
Answer	According to Act on Atomic Energy, Section 19/A: “(1) Nuclear facilities shall be required to pay a regulatory fee to HAEA. (2) The annual regulatory fee shall be established: a) for nuclear power plants and research reactor by multiplying the nominal heat capacity (MWth) by the calculation base,… b) for temporary storage facilities for spent reactor fuel by multiplying the number of spent fuel containers in storage at the end of the previous year by the calculation base,… (3) Nuclear facilities shall pay the applicable portion of the annual regulatory fee quarterly, by the fifth day of the quarter at the latest.” The regulatory fee paid by the licensees covers about three-quarters of the annual budget of the Authority.		

Seq. No 58	Country Finland	Article Article 11.2	Ref. in National Report chapter 3.4.2
Question/ Comment	Firstly, we note that PAKS NPP has excellent training services and facilities including full-scope replica simulator facility and maintenance training facility available at site as described in paragraph 3.4.2. Could other NPP's with similar type reactors benefit from these training facilities and services?		
Answer	In spite of the fact that Paks NPP Ltd. indeed has excellent training facilities for both the operational and maintenance personnel, there are trivial limitations against their use by other NPPs with similar reactor design. The full-scope replica simulators are meant to respond to plant specific designs, thus however minor plant design differences among the VVER-440s are, the use of the Paks simulator by other CRO staffs is practically impossible. The Maintenance Performance Improvement Center holds an outstanding collection of VVER-440 (V213) main components which can be used by other NPPs for demonstration and train the trainer or training development services. The plant offered its services to other VVER plants but there was no request from their side to use the services of the Paks NPP		

	maintenance training facilities.		
Seq. No 59	Country Germany	Article Article 11.2	Ref. in National Report p. 35, 3.4.1
Question/ Comment	Who is responsible for inspector's examination within the HAEA in order to guarantee the objectiveness (e.g. competent persons from outside)? Does the examination also include simulator and safety management training?		
Answer	The HAEA NSD Training official is responsible for the training and education of the HAEA inspectors. The newly entering staff have to participate in a two-year-long training, while they get familiar with - the internal procedures, - the regulatory procedures, - the operation of the different licensees, - laws and regulations in the area of nuclear energy. The training program includes theoretical and practical studies at the authority, at the licensees (including courses held for reactor operators), and abroad. After the training period the new inspectors have to take a final exam in front of a board of examination, which includes senior experts with great experience and which is headed by the head of HAEA NSD. Objectiveness is guaranteed by the head of the examination board. After the exam the new inspectors may work individually. There are also annual training plans to ensure the maintenance of the acquired knowledge At the exam the candidates also have to answer questions regarding the safety management. At the exam there is no special simulator task, but the inspectors are trained at the NPP's full-scope simulator as well.		
Seq. No 60	Country Germany	Article Article 11.2	Ref. in National Report p. 35, 3.4.1
Question/ Comment	also to Article 8.1 Does the training of HAEA inspectors also include simulator and safety management training?		
Answer	Yes, the training of HAEA inspectors also includes training at the NPP's full-scope simulator as well as safety management training. After recruitment the new inspectors have to participate in a two-year-long introductory training, which also includes the above mentioned areas. The training is typical for each new inspectors, it is based on the previous qualification of the employees. After the training program the new inspectors have to take a final exam in front of a board of examination, which includes senior experts with great experience. After the exam the new inspectors may work individually. Annual training plans are elaborated in order to ensure the maintenance of the acquired knowledge level. The annual training plan is divided into 3 chapters. The first chapter describes the training program for the new entries. In the second chapter the refreshing programs are listed. The third chapter describes further education training courses. Among the refresher training courses there are biannual simulator training, and annual safety management training.		
Seq. No 61	Country Austria	Article Article 12	Ref. in National Report
Question/	Which root cause analysis of events in NPPs have taken place including thorough		

Comment	consideration of human error? What lessons on human factor and deterioration of safety culture were learnt from the incident on 10 April 2003? Which training measures were introduced as consequence of this incident?
Answer	Paks Root Cause Analysis Procedure (PRCAP) was originally an adaptation of the Human Performance Investigation Process (HPIP) of the US NRC and the safety management factors in the Management Oversight & Risk Tree (MORT) of the US Department of Energy. Nevertheless, significant modifications and amendments have been made which reflect the all-round comprehension of the RCA methods currently used in the world and the specific requirements for RCA at Paks NPP. The following deviations related to human factor and safety culture have been identified during the investigation of the incident on 10 April 2003: • The reviewers did not follow the related requirements during the review process of the design. So the weaknesses of the cleaning tank design have not been explored. • The Paks NPP overall responsibility in the area of nuclear safety has not been enforced due to the weak or careless management processes and missing requirements • Paks NPP personnel involved in the fuel assemblies cleaning process have not been properly trained. • Root cause: Short time advantages and production have got priority over safety considerations in the 90-es when decisions were taken on the necessity of SG feed water collector replacements. A comprehensive training program has been fulfilled including presentations of the lessons learned for different levels of the employees from workers to managers. Classroom trainings for the shift personnel have been carried out. Beyond these as a long-term measure, a program for the development and improvement of the NPP organizational processes has been launched.

Seq. No	Country	Article	Ref. in National Report
62	Austria	Article 12	

Question/ Comment	Are there fixed schedules for human factor inspections ?
Answer	HAEA performs a comprehensive inspection program that includes as one of the main areas the Human Factor field. Inspection is carried out once in every two years in this area. On the other hand - based on the results of investigation of safety related events - HAEA evaluates yearly the human performance. The operators have to renew their license in every three years by oral exams and every 6 months on the simulator. As a part of the inspection activity the representative of HAEA, as a member of the examination board, takes part on the exams.

Seq. No	Country	Article	Ref. in National Report
63	Japan	Article 12	P.38/L.18

Question/ Comment	It is reported one of the strategic goals of the company is to extend the service life of the four units of the NPP by 20 years beyond the design lifetime. Could you show the details on the followings? (1) law that prescribes the procedure for lifetime extension of nuclear power plant (2) regulatory requirement and acceptance criteria for lifetime extension of NPPs (3) competence and resources of regulatory body

Answer Ad. (1) and (2)

The legal framework for regulating nuclear safety relevant to lifetime extension is as follows: Atomic Act (Act CXVI of 1996) and the related 108/1997 Korm. Governmental Decree. They address the issue of lifetime extension as described below:

• According to the atomic act a licence (among them the operating licence) may be granted for a defined or undefined period of time, as well as subject to certain stipulations. The licence granted for a defined period may be extended when so requested.

• The Governmental Decree issued for the execution of the act clarifies that the issuance of operating license could mean the extension of the designed lifetime.

• According to the decree, so as to extend the design lifetime of the NPP units, no later than four years before the expiration of design lifetime, the Licensee shall, submit a program to the regulator, which schedules the establishment of the conditions of the operability beyond the designed lifetime. The regulator inspects the program and its implementation.

• Licensing of operation beyond the design lifetime takes place through the new operating license issued before the end of design lifetime upon the application of the Licensee. Within the procedure assessing the application the regulator considers the results of the program and its inspection findings.

Detailed regulations

Within the Hungarian nuclear regulation system the detailed prescriptions are involved into the Nuclear Safety Regulations. The regulations were issued as the appendices of the mentioned governmental decree. There are six volumes of these regulations from which the first four is related to the NPP (the other two address the research reactors and spent fuel storage facility). This four volume divide the nuclear requirements as follows:

Volume 1: Regulatory procedures,
Volume 2: Quality assurance,
Volume 3: Design
Volume 4: Operation

Originating from this fact it concludes that the regulation of different issues (for example lifetime extension) is addressed by more than one volume.

The regulation divides the lifetime extension procedure into two stages:

c) program for lifetime extension,
d) new operating license.
a) Program for lifetime extension

According to the regulation the safe operation shall be continuously maintained during the preparatory phase and during the operation beyond the designed lifetime (OBDL) in accordance with the laws and regulatory prescriptions of legal force. The problems arising from the actual operation shall be handled within the valid operating license. During the OBDL the necessary safety margins, considered by the safety analysis, shall never be consumed, not even with reference to the approaching of the end of licensed lifetime. The activity aiming at maintaining the technical conditions of the safety SSCs shall be launched and continuously performed already within the designed lifetime; additionally the efficiency of this activity shall be systematically supervised and evaluated. The determination of safety improving measures, deriving from the modern international requirements, shall be carried out within the frame of PSR and not for the lifetime extension issue.

Requirements for the program aiming at establishing the conditions for lifetime extension:

- For establishing the conditions of lifetime extension and for the justification of operability the Licensee shall prepare a program. The program and a description of its time-proportional implementation shall be submitted to the regulator no later than four years before expiration of the design life. The program can be submitted for one or more units of the same plant. In the substantiating documentation at least 20 years of operating experience shall be considered. The regulator inspects the program and its implementation (and checks for any discrepancy that could prevent licensing of lifetime extension).
- All modification and fixing activity shall be performed within the frame of the valid operating license and not in the program.
- The program shall be based on the requirements for the application of the new operating license. Here the fulfilment or status of fulfilment or the activity (with schedule) planned for the fulfilment of that requirements should be demonstrated.
- The program shall contain the planned duration of OBDL.

b) Operating license (OL)
Licensing of lifetime extension is performed in the new OL, upon the application of the Licensee to be submitted 1 year before the expiration of the lifetime. Validity: until defined time period if all conditions are fulfilled. In the OL application it should be demonstrated that:

- appropriate scoping of SSCs necessary to safe OBDL is performed;
- relevant ageing mechanisms are addressed;
- the condition of relevant SSCs are surveyed, efficiency of the former ageing programs are evaluated, new ageing management aspects and requirements are elaborated;
- scope of time limited ageing analysis (TLAA) involved in lifetime extension is determined, former TLAAs are re-evaluated and their validity is checked;
- the FSAR is actualized;
- necessary modification of operating conditions and limits are surveyed and substantiated;
- relevant documents (operating limits and conditions, maintenance policy, symptom-based emergency operating procedures, other emergency procedures, emergency response plan) are surveyed and their modifications necessary to lifetime extension are justified.
- Upon the above activities it is ensured that during extended lifetime the safety function are fulfilled at the desired reliabilty, the safety analysis covers the possible operating modes and the operating limits and conditions are in harmony with lifetime extension requirements.

The followings shall be attached to the OL application: actualised FSAR, modified version of the above documents, the necessary special authority contributions. Background documentation to the substantiating documents shall be submitted upon further regulatory request.

Re-licensing of operating and other licenses expired at the end of lifetime

Conditions for the issuance of the operating license: the temporary storage or final disposal of radioactive wastes and spent fuel shall be ensured in harmony with the international expectations and experience. The valid operating license is precondition; maximal length is the operating license of the unit. In the application the followings shall be demonstrated:

- The operation is in accordance with the approved safety analysis.
- The inspection, manual and emergency documents and procedures are appropriate for safe operation.
- Necessary initial data for condition monitoring of the SSCs are available.
- Safe operation is ensured fulfilling the operating limits and conditions.
- Technical and administrative conditions are ensured for long term safe operation, the financial resources performing long term maintenance and development of safety are available, the possible reasons for cancellation of the license are eliminated.
- The documents and contributions needed to OL are also parts of this application.

Guidelines relevant to lifetime extension

Besides the legally binding requirements the regulator has the possibility to issue not legally binding requirements. However these regulatory guidelines has important role in the system of regulations, because if the Licensee would like to deviate from the given guideline than it shall be justified that the applied method is more or at least equally conservative than the one of the guideline. This method shall be well substantiated.

The system of guidelines follows the structure of the Nuclear Safety Regulations; that is all of them are attached to one of the volume of the NSRs. So for example concerning ageing there are 4 guidelines in which four different aspects of requirements are address. In the guidelines the requirements of the NSRs are explained in details or the method of meeting the given requirement is formulated.

Concerning lifetime extension the following guidelines were already issued:

Maintenance

1.19 Inspection of the efficiency of the maintenance program of the nuclear power plant

4.8 Nuclear power plant maintenance program and maintenance efficiency monitoring

Ageing

1.26 Regulatory Inspection of the Ageing Management Program

2.15 Quality Assurance in the Ageing Management of Nuclear Power Plant Equipment

3.13 Consideration of Ageing during Nuclear Power Plant Design

4.12 Management of Ageing During Operation of Nuclear Power Plants

Equipment qualification

1.27 Regulatory control over equipment qualification and preservation of the qualified status

3.15 Equipment qualification requirements during the design of nuclear power plants

4.13 Equipment qualification requirement for operating nuclear power plants

Additionally the two most relevant guidelines are under issuance. These guidelines directly address the lifetime extension. The titles and numbers will be:
1.28 Requirements for the scope of the lifetime extension licence application
4.16 Conditions of operation licence renewal of nuclear installations

Ad. (3)

HAEA NSD will train those persons from the staff who will be involved in the lifetime extension process and the TSOs provide support in the assessment of the application. The results of the R&D activities in the area of lifetime extension have ensured an adequate background for establishment of the legal framework.

Seq. No 64	Country Slovenia	Article Article 12	Ref. in National Report section 3.5.5, p 38

Question/ Comment

Subsection 3.5.5 describes that for enhancing safety culture the NPP has performed the assessment of the actual level of safety culture on three occasions.
When were the safety culture assessments performed?
By comparing of these three results, what was the trend?
For the incident in April 2003, a major root cause was weak safety culture of the NPP – was this observed also by these assessments (does Unit 2 show worse results than other Units)?

Answer

The safety culture assessments mentioned in the question were performed in 1995 and 1999 for the general employees of the company. Some positive change in the values of numerical indicators was observed. However, those indicators proved not to be very suitable to really evaluate the changes. The comments made by the interviewed people and the defined corrective actions based on those comments proved to be more beneficial in the process of safety culture improvement. A similar assessment was performed for the managerial staff in 2000. Looking back to the results of the surveys some signs of deficiencies could be seen, however, those signs did not seem to be very significant at that time. Furthermore, in order to eliminate those deficiencies corrective actions were defined. As mentioned in answers to other questions, after the incident in April 2003 the safety culture assessment methodology was revised.
The results of the safety culture assessment are not unit specific. They characterise the safety culture of the Paks NPP Co. as a whole. Therefore unit 2 cannot be separated from the others. The entire plant with four units have a common organisation and management system.

Seq. No 65	Country Austria	Article Article 13	Ref. in National Report

Question/ Comment

With regard to the incident on 10 April 2003 it seems to be obvious that the quality assurance standard of the NPP should be improved. Is it planned, - and if yes, under which timeframe - to adopt a quality assurance standard, like e.g. EN-ISO 9000, in the NPP?

Answer

The requirements of ISO 9001:2000 and Paks NPP Quality System (2nd Volume of Nuclear Safety Regulations (NSR)) have been compared. The plant has determined the extra requirements specified by ISO which are progressive and seem to be reasonable to be taken into account by the Paks NPP.
The Quality System of the NPP is in accordance with the 2nd Volume of the NSR. As a result of the ongoing development of the management system the Quality

System shall incorporate the extra requirements by the end of 2005. In general it can be stated that the system also fulfils the recommendations of the IAEA safety series document 50-C/SG-Q.
It should also be mentioned that regarding the Environmental Management System the Paks NPP has the qualification ISO 14000.

Seq. No	Country	Article	Ref. in National Report
66	Austria	Article 13	

Question/ Comment	A design mistake with the cleaning vessel caused the incident on 10 April 2003 . Which measures have been taken to implement the responsibility of the licensee to hire only subcontractors who have an appropriate quality assurance system? In which way did the licensee check the subcontractors´ quality assurance? How was it possible that the HAEA's QA system did not detect the design fault – despite its ISO certification?
Answer	When auditing the subcontractors of the NPP (and during work performance on site as well) the documented qualification of employees performing activity shall be checked individually. The necessary legal authorization shall also be checked for the employees performing design, expertise and technical inspection. The certification of subcontractors' quality system is carried out by Paks NPP based on written procedure. Generally – also in the case of FRAMATOM ANP – the certification process is started by preliminary check of the subcontractors' quality system (review of third party certificates, quality documentation) and followed by on-site audits. During certification process we make sure that the subcontractors' quality system is in conformance with related requirements of Paks NPP. The ISO system helps to execute the required workflow, which was designed under the preparation phase of the ISO system. The implemented Quality Management system applies requirements also against the activities of the consecutive working steps of assessment. The QM doesn't go to such details, whether in the given instance of the cleaning tank case, how deeply should the licensing staff dig into thermal-hydraulic calculations. The QM prescribes the requirement that there must be evidences to decide on the acceptance. Since the designer company appended thermal-hydraulic calculations according to the safety obligations, the evidence was present. The design verification is an engineering process, which in the first place is not the task of the authority personnel but of the industry. The licensing staff might have been suspicious about the results of the thermal-hydraulic analysis and definitely made a mistake when missing the opportunity to rise questions about the adequate number of measuring instruments, first of all, for temperature measurements. An important issue is that the regulatory personnel is not necessarily able to repeat every design and analysis step what were done by the design or engineering company specialists. The well defined workflows and checking mechanisms are expected from the design, manufacturing and operating companies, the authority workflow is oriented more to the existence of the evidences, whereas the content of the evidences should be interpreted by the authority staff.

Seq. No	Country	Article	Ref. in National Report

67	Canada	Article 13	3.6.5, page 43

Question/ Comment
Subsection 3.6.5 of the report describes the areas of inspection that the Authority conducts to check the licensee's quality assurance system. Please explain how these activities verify the ability of the licensee to select and manage contractors.

Answer
The requirements for the contractors of the Paks NPP are described in the Vol. 2. of the Nuclear Safety Regulations (Points 3006-3008. of NSC Vol. 2. and points 56-60. of its Attachment). Procedures of the QA system of the Paks NPP – that are checked regularly by the Authority – regulate the qualification processes of the contractors in details. These processes are executed by the representatives of the Paks NPP assigned for special fields together with the permanent auditors of the personnel of the Paks NPP. On the basis of the Annual Plan of the Utility the Regulatory Body selects some activities and contractors and its representative auditor takes part in the qualification process of the selected items. The auditor has the right of veto. The Regulatory Body concentrates on the qualification of contractors working on systems of 1. and 2. safety. On a yearly level the Regulatory Body takes part in about 83 % and 43 % of the qualification processes belonging to the safety classes 1. and 2., respectively. The regulatory auditor among others checks
- the efficiency of the plant auditors,
- the existence and completeness of the QA system of the contractors,
- the effectiveness of the QA systems,
- references in the nuclear industry.

Seq. No	Country	Article	Ref. in National Report
68	France	Article 13	§ 3.6.5 - p. 44

Question/ Comment
Could Hungary indicate which improvement measures were requested by the Authority in the licensee quality assurance organisation?

Answer
The regulatory body has made a resolution that the operator should prepare an integrated corrective action plan at the second half of 2003. One of these actions is the program of organizational development in which it is required to assess the functions and operation of the NPP's organization including the quality assurance organisation. The regulatory body continuously follows the performance of the corrective actions plan and so far no other regulatory actions were necessary. Besides the Authority mandated actions the IAEA expert mission has recommended and suggested some specific actions for the quality and safety management area. (For example: the control and assessment of the contractors, internal inspection program)

Seq. No	Country	Article	Ref. in National Report
69	Korea, Republic of	Article 13	3.6.4 (p41)

Question/ Comment
In relation to paragraph 3.6.4, 'Quality Management System of the Nuclear Power Plant System', it is stated that an indicator system is used to assess the correct functioning of the QA System.

1. What are the elements in the indicator system?
2. Who assesses the QA system ? Is it done by an internal or external organization?
3. Please explain in more detail how the indicator values are used to improve quality management system.

Answer
1. Development of Paks NPP self-assessment system was one of the recommendations by the OSART mission in 2001. Thus the NPP has started to

develop a Safety Indicator System based upon traditionally used WANO indicators together with a comprehensive system of performance indicators. At present the key processes and their indicators are defined. By the end of this year each process shall have its own performance indicator.

2. The Paks NPP Quality Assurance System assessment is done by traditional review tools such as internal audits carried out by the NPP personnel and comprehensive reviews performed by Hungarian nuclear regulator.
The systematic self-assessment was introduced at Paks NPP in 2002. The performance indicators support the objectivity of the self-assessment. The results of self-assessment are summarized annually. The key elements of the self-assessment and the trends of the system development are also defined.

3. The values of the safety indicators are evaluated quarterly by the members of the Safety and QA Management Committee and in case of deterioration corrective measures have to be defined. The indicators of the key processes are evaluated yearly with the same objectives i.e. in order to define corrective actions if needed.

Seq. No	Country	Article	Ref. in National Report
70	Korea, Republic of	Article 13	3.6.5
Question/ Comment	Does the regulatory authority perform a system or process audit regularly? If yes, how frequently does the authority audit each operating organization of NPP? The second paragraph of Sub-article 3.6.5 indicates that the audits are carried out by internal auditors. Does it mean the authority uses qualified auditors of the utilities?		
Answer	In the framework of the overall inspection system of the regulatory authority (regulated by an internal procedure) there are at least three audits affecting the QA system of the utility every year. The main areas can be: maintenance, education, technical aspects of radiation protection, waste management etc. Besides the special QA aspects of the actual areas basically the management and the relevant QA organisation unit are inspected. The same areas are inspected with a 2-3 year periodicity. If an audit is focused expressedly on the QA system of the utility the regulatory group is lead by a highly qualified QA expert having an international certification.		

Seq. No	Country	Article	Ref. in National Report
71	Slovakia	Article 13	3.6
Question/ Comment	Please explain how do you assure the quality of the regulatory work. Do you measure effectiveness/efficiency of regulatory work? At which success? Which criteria do you apply? Are the measured/evaluated trends positive?		
Answer	HAEA has implemented a quality management system certified according to ISO-9001-2000 standard. As a part of this system an effectiveness/efficiency measuring procedure has been introduced. The characterising indicators are quantitative, represent the quality of the regulatory activity and can be used for trending. The evaluation of the results is under development.		

Seq. No	Country	Article	Ref. in National Report
72	Slovenia	Article 13	section 3.6.5, p 43
Question/ Comment	Subsection 3.6.5 states that the areas of the operator's quality assurance system regularly inspected by the Authority are as follows (in connection with the control		

activities):
- structure of the organisation,
- training and qualifications of staff,
- documentation,
- treatment of non-conformity.

Does the Authority regularly inspect the area of inspection and testing for acceptance?

Answer

In the text of the Hungarian National Report the areas of the operator's quality assurance system regularly inspected by the nuclear safety authority are divided into two categories: "activities connected with control" and "activities connected with execution". The question refers to the first one, whereas this categorisation in the text is based on an inaccurate translation, as originally the first was the group of management activities and the second the group related to performance (using the terminology in the QA Code of the IAEA [50-C/SG-Q]). According to this subdivision the area contained in the question (inspection and testing for acceptance) is related to the performance rather than to the management activities. The activities related to inspection and testing for acceptance in practice are mainly either a part of the area "maintenance and repair work", or a part of the area "modifications" (after their implementation). Most naturally these areas are regularly inspected by the HAEA NSD - as it is also expounded in the National Report.

Seq. No	Country	Article	Ref. in National Report
73	Austria	Article 14	

Question/ Comment

What are the results of the Level 2 PSA? When will the Level 2 PSA be expanded to include seismic events?

Answer

The main objectives of the level 2 PSA study were: (1) to provide a basis for the development of plant specific accident management strategies, (2) to provide a basis for the plant specific backfit analysis and evaluation of risk reduction options, and (3) to provide a basis for the resolution of specific regulatory concerns.

The scope of the PSA-2 currently covers the full power PSA-1 with internal initiators, fire and flooding hazards and PSA-1 for the low power and shut down states. Beyond the different reactor risk studies the review of all other hazardous non-reactor systems on the site containing radioactive materials have been performed, like the spent fuel storage, the hydrogen burning systems in the main and other buildings, etc.

As for the quantitative results, the annual frequencies of large radioactive releases for different predefined severity categories were calculated. The severity was correlated to the amount of caesium released. The frequencies of the presumably most severe accidents (high energy reactor pressure vessel damage, containment by-pass) are low, around 10-7 1/year. The second and the third severity categories (releases higher than 1000 TBq Cs) have together a frequency of about 5•10-6 1/year for the full power operation without any accident management assumed. According to the estimations the release frequencies in accidents in the shut down states and in accidents of the spent fuel storage pool are relatively high.

The risk reduction capability of different accident management possibilities has been assessed. The accident management program is submitted to the regulator, the review process is ongoing. This program comprises hydrogen treatment by using recombiners, flooding of the reactor shaft for either the external cooling of the reactor vessel or for protecting the basement from melt through, filtered venting, prevention of the rector shaft door damage as mitigative measures. A number of other improvements, mostly preventive measures are suggested to decrease the accident initiating frequencies in the shut down states and of the spent fuel pools.

Upgrading suggested by the level 1 seismic PSA study is close to finalisation. The recent level 1 PSA model will then be modified and updated. That new model would be the basis for inclusion into the level 2 PSA model. The corresponding results will not be available before 2007.

Seq. No 74	Country Canada	Article Article 14	Ref. in National Report 3.7.1, p44;3.7.5 p46

Question/ Comment: What are the criteria and guidelines established by the Authority to its staff for the review of the licensee's Periodic Safety Report?

Answer: The guideline for preparation the PSR was issued as a resolution by the authority. The content of this guideline was determined to account for all important safety issues and was based on the Nuclear safety Regulations and on the IAEA SG on the subject. (50-SG-O12). The review carried out by the utility staff was based on the guideline. The information given by the Report had to be compared to the other documents that were available for the authority. Furthermore, the personal knowledge of the people doing the review was very important in this activity.

Seq. No 75	Country Canada	Article Article 14	Ref. in National Report 3.7.5, page 47

Question/ Comment: The report indicates that in dispositioning the PSR, "the Authority prescribed 65 improvement measures, 15 are of such whose delayed accomplishment would suspend the validity of the operating license."
What were the success indicators established to judge the accomplishment of these 15 improvement measures?
What were the factors considered to set deadlines to accomplish these 15 improvement measures?

Answer: The safety improvement measures and their deadlines were proposed by the utility in the Periodic Safety Report. All deadlines were negotiated by the experts of the utility and the authority, and they were fixed according to a common agreement. This is not to say, that there were no differences between the utility and authority opinions, but these conflicts were eliminated during the discussions.

Seq. No 78	Country Netherlands	Article Article 14	Ref. in National Report 3.7.4, page 45-46

Question/ Comment: In 3.7.4 on page 45-46 it is stated:
"Evaluation of the seismic resistance and the implementation of the prescribed reinforcements were fully completed by the end of 2002. In 2002 the complex testing of the cool-down technology, elaborated in order to implement seismic safety, took place and the Authority granted the license for its future application. As from 2003 the seismic safety of Paks NPP meets the requirements prescribed by the International Atomic Energy Authority."
What were the most important reinforcements that were implemented as a result of the seismic safety review?

Seq. No	Country	Article	Ref. in National Report
76	Korea, Republic of	Article 14	p45-46

Question/ Comment

(Article 14, 3.7.4 Seismic safety, pp.45-46)
It is stated that the site evaluation, and the evaluation of seismic safety and implementation of reinforcement and modifications within the "easy-fix" project were carried out for Paks NPP.
1. What were the earthquake levels (horizontal and vertical peak ground accelerations) and design ground response spectra in the original seismic design and the seismic safety evaluation?
2. What was the seismic safety evaluation methodology in "easy-fix" project?
3. What were the details of the reinforcement and modifications of the NPP?
4. What were the details of inspection of the stability of the basement in p.45?
5. It is stated that the seismic safety of Paks NPP meets the requirement prescribed by the IAEA. What requirements were met? How does it mean the seismic safety meets the requirements?

Answer

Question 1: Originally, the Paks NPP was not designed for earthquakes loads. The intensity MSK 6 was assumed for the site seismicity. The ground peak acceleration correlated with this intensity was found to be negligible in the design. It complied with the regulation valid at that time (sixties) in the Soviet Union.

That was one of the reasons for the re-evaluation of site seismicity and upgrading of the plant started in the mid eighties.

Questions 2 and 3:
See Support Document "Netherlands' question Seq.No. 2"

Question 4: The stability beneath the foundation was carefully analysed. This was based on the geotechnical investigation of soil in order to assess the soil liquefaction hazard. The probability of soil liquefaction was found to be less than 10-4, therefore the liquefaction was excluded from the design base. (See the reports: Site Investigation of Site Response and Liquefaction Potential, Interpretative Report, Ove Arup & Partners, London, October 1995. and Site Investigation of Site Response and Liquefaction Potential, Final Report, ISMES, Bergamo, August 1995.)
Although the liquefaction was not included into the design basis, the dynamic settlement and stability of building was checked taking into account the variability of soil parameters beneath the buildings.
The soil liquefaction has been studied also in the frame of seismic PSA for the earthquake levels with very low probability.

Question 5: The requirements of the IAEA in relation with seismic safety are defined in two aspects:
• siting, acceptability of site, definition of design base (SL-2) earthquake characteristics
• design and qualification
These requirements might be extended to the preparedness of the operators to the earthquakes.
The IAEA requirements were formulated in the documents:
• IAEA Safety Series No. 50-SG-S1 (Rev. 1) Safety Guide, “Earthquake and

Associated Topics in Relation to Nuclear Power Plant Siting", 1991.
• IAEA Safety Series No. 50-SG-D15 Safety Guide, "Seismic Design and Qualification for Nuclear Power Plants", 1992.
These regulatory documents have recently been revised.
Concerning siting: The site survey and hazard assessment complies with the relevant requirements of the above mentioned IAEA guidance, e.g. the existence of capable fault is excluded, the site specific design base earthquakes is defined, the soil proneness of the liquefaction was studied.
Concerning the design and qualification requirements: The Paks NPP after upgrading meets the requirements defined in the IAEA guidelines, however the specific conditions and assumptions valid for the re-evaluation and upgrading of operating plant were taken into account. These specific assumptions and conditions were defined later in the IAEA guidance document "Seismic Evaluation of Existing Nuclear Power Plants, Safety Series Report No 28, IAEA, Vienna, 2003" which is applicable for seismic re-evaluation and upgrading of operating nuclear power plants. After implementation of the seismic upgrading project Paks NPP complies with the IAEA requirements defined in the documents mentioned above with those specific assumptions which are applicable for operating NPPs.

Brief description of the easy-fix project

As the preliminary results of the site seismic hazard re-assessment showed that the seismicity should be higher than it was assumed in the design, the easy-fix project was launched (1993). The program was completed in 1995. The intention was to implement all measures which were found to be urgent and also feasible on the basis of preliminary and very conservatively defined reference level earthquake.

The work was started with definition of the scope of easy-fix program. The scope of easy-fix program is a subset of the list of systems structures and components relevant to seismic safety.

For the qualification purposes the NUREG/CR-0098 soft site, median response spectrum was selected for the maximum horizontal acceleration of 0.3 g. Meanwhile for design of the fixes the US NRC Regulatory Guide 1.60 response spectrum was used for the value of 0.35 g.

Walkdown screening: using NUREG 1407 for IPEEE walkdowns to check compliance with the walkdown guidelines and screening criteria, anchorages, weak links in load paths, seismic interactions.

The qualification was made using procedures based on seismic margin method and walkdowns. Qualification methodology: NP 6041, GIP, simplified evaluation, basically EQE and Westinghouse experience.

Seismic load and floor response spectra: 3D structural response was calculated for the conservatively assumed GPA and Reg.Guide 1.60 spectra anchored to 0.35 g, and scaled to 0.3g for the evaluation purposes.

The main findings were as follows:

Mechanical equipment: rugged, some anchorage improvement needed;
Electrical and I&C: anchorage problems, mainly top bracing added, fixes inside needed in some cabinets, functionality check and relay chatter analysis are separate, non easy-fix tasks
Cable trays: static over-loading, new supports
HVAC ducts: additional supports, ventilators anchorage improved for lateral loads
Seismic interactions: mainly the masonry besides of the safety related I&C was identified as easy-fix items, other interactions were also identified and evaluated, taking into account also the results of ongoing HELB project

total number of items in the preliminary SSE list:	10184 for 4 units	improvements
total number of easy-fix items	5507	
mechanical equipment	202	anchorages
electrical equipment	465	anchorages
cable trays	2498	anchorages
I&C (cabinets, racks)	2061	anchorages and top bracing
brick walls	281	Steel frame fixes
total amount of steel for fixes	445 tons	
Safety related batteries replaced and properly fixed	yes	

The seismic evaluation and upgrading programme implemented after the site seismic hazard re-evaluation had much broader scope as the easy-fix project.

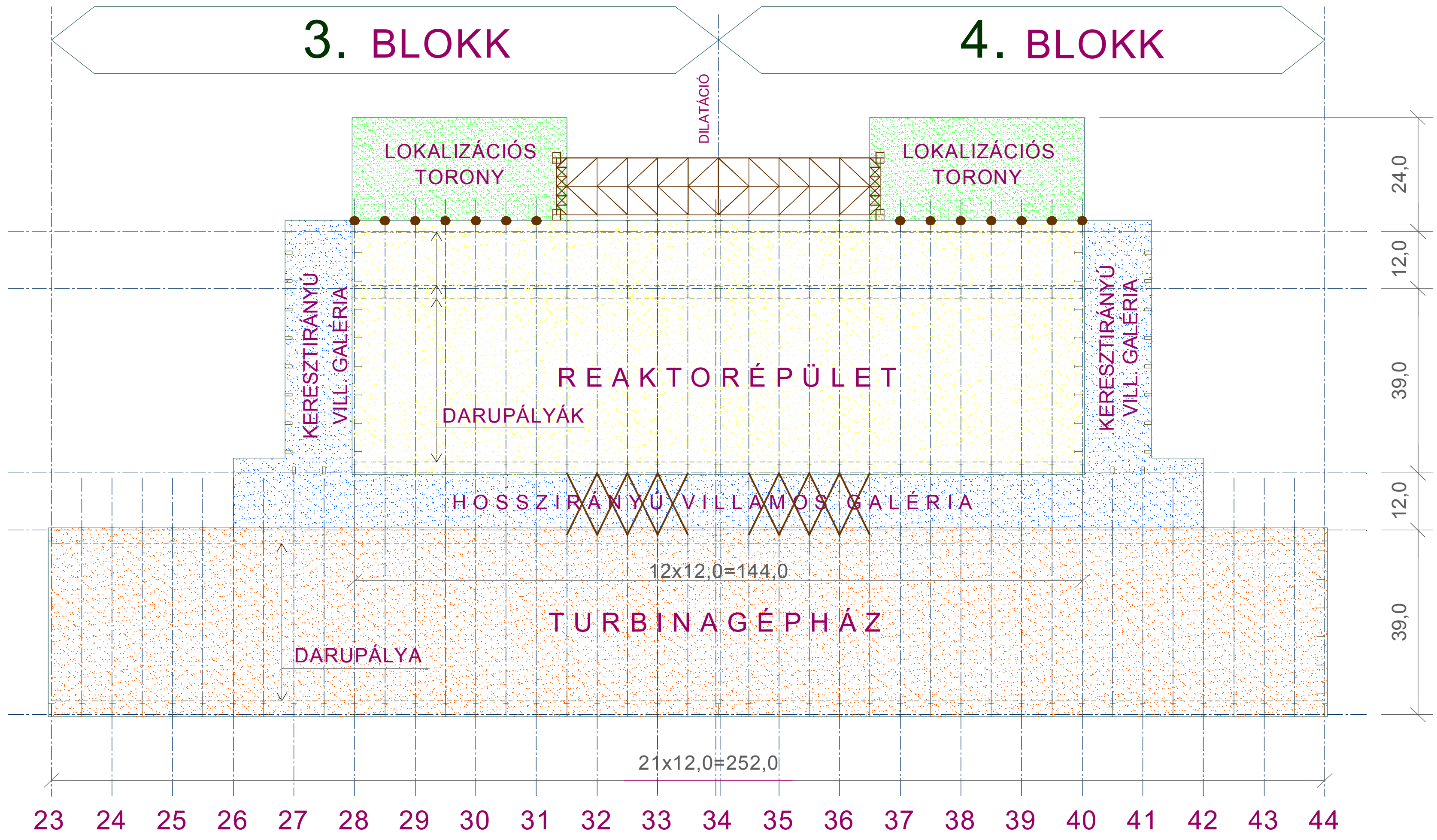
The newly constituted design base earthquake (DBE) characteristics had been used for the evaluation and design of upgrades. The maximum horizontal ground peak acceleration of the DBE is equal to 0.25 g, and the vertical one 0.20 g. The site specific uniform hazard response spectra were defined and the local soil conditions were taken into account.

The extent of the seismic evaluation and upgrading project based later on the final results of site seismic hazard re-evaluation is illustrated in the table below:

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

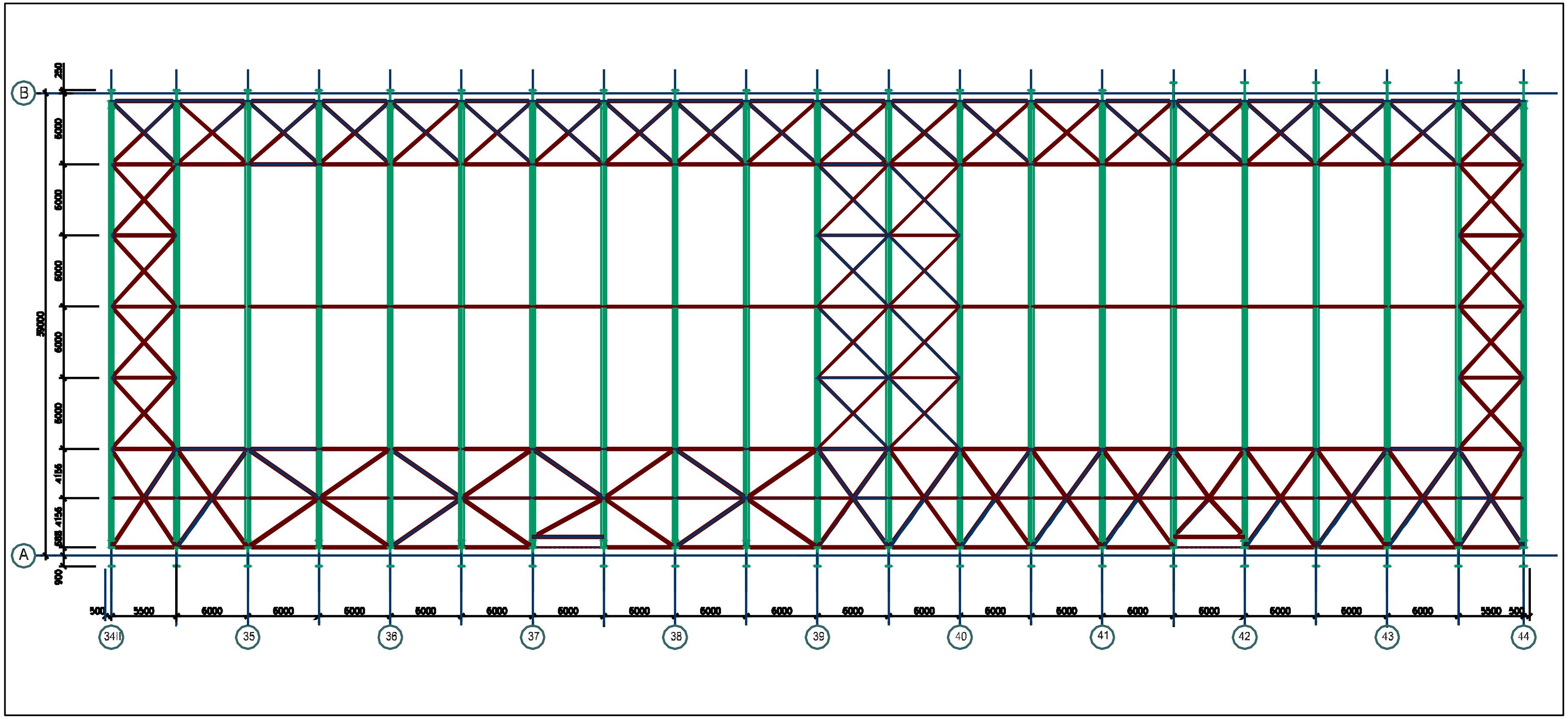
Typical examples for easy-fixes, top bracing of cabinets and fixes for masonry walls:

Additional X-bracing in the reactor building



Anchoring of the steel-frame structures into the rigid reinforced concrete block of reactor building

The bridge between the two localisation towers. They are also for the transfer of transversal loads to the rigid reinforced concrete part of the buildings.



Fixing of the roof-belt (red lines indicate the new elements)

Visco-dampers on piping and equipment

Seq. No	Country	Article	Ref. in National Report
77	Korea, Republic of	Article 14	3.7.4 , p46

Question/ Comment (Article 14, 3.7.4 Seismic safety, p.46)
It is stated that the earthquake alarm and protection system currently operates off-line in order to prevent unit shutdowns triggered by false signals. However, in general false signal from vibrations other than earthquakes has nothing to do with the off-line operation. What do you mean by the off-line operation and false signals?

Answer A seismic instrumentation has been installed on each unit of the plant in 1993.
The instrumentation consist of seismic switches mounted on the base mat, sensitive accelerometers registering the response at the characteristic points of the structure, appropriate data collection system and voting logic. Two free field stations are installed at the plant too.
The hardware would allow adaptation of different principles of shut-down of the plant in case of an earthquake.
In 1993 it was also recognised that the probability of spurious scram due to ambient vibrations or impacts or failure of instrumentation is too high, consequently the automatic scram based on the acceleration trigger level crossing is disadvantageous and could not be justified by safety reasons.
After extensive study of international practice the cumulative absolute velocity (CAV) criteria of continuous safe operation (and for the cases of exceeding OBE) had been adapted. The necessary data acquisition and evaluation procedures and software have been developed. An emergency procedure exists which determines the operator activity after the earthquake. A comprehensive guidance was elaborated for assessing the post earthquake situation at the plant. The applicability of CAV criteria for the continuous safe operation is based on results of capacity evaluation of the SSC as well as the experience data behind of the response spectrum and cumulative absolute velocity limits.
It means, that no automatic reactor scram will be activated, if an earthquake occurs.
There is a specific condition valid for Paks NPP to be considered. According to the Hungarian regulation, reactor cool down and heat removal unlimited in time have to be ensured after an earthquake. These requirements implicitly mean that essential parts of the plant have to be operable after an earthquake and that those part which are not needed for the heat removal (and not fixed) have to be separated from the operable and fixed ones by quick-closing valves. This separation will happen automatically based on the (OBE) acceleration trigger level crossing. However this separation itself does not disturb the continuation of normal operation. Therefore these conditions might be kept, while the signal records are evaluated, the CAV criteria and those of exceeding of OBE are defined.

Seq. No	Country	Article	Ref. in National Report
79	Netherlands	Article 14	3.7.6, page 48

Question/ Comment In 3.7.6 on page 48 it is stated:
"The annual average probabilities of core damage originating from the assumable incidental situation as a consequence of events including all operating states (full power operation, shutdown states during refuelling or overhaul), of an internally caused system- or equipment failure, inadequate human interactions, internal origin fire and flooding were, in 2003, for the four units in sequence: 3.8x10-5; 3.3x10-5; 4.4x10-5; 3.6x10-5.Paks NPP Ltd performed the

seismic assessment of the selected reference unit and it determined the value of the anticipated core damage frequency. By virtue of the significant similarity and architectural identity of the units, this value is valid for the other units as well. The calculated average value of core damage frequency of a unit of the nuclear power plant originating from the accident scenario postulated as a consequence of an earthquake is 2.87x10-4 per a year."
The core damage frequency as a consequence of an earthquake is calculated to be one order of magnitude higher than all possible accidents with an internal origin, including fire and flooding. Are any measures considered to further reduce the frequency of the earthquake scenario?

Answer The seismic PSA studies have been started in the final phase of the seismic upgrading program. The seismic PSA performed for the plant before essential seismic upgrading implemented would identify large number of trivial contributors of risk. The objectives of the seismic PSA was

- to quantify the safety of the plant in case of an earthquake
- to confirm the procedure of cool-down and heat removal
- to judge about contribution to the safety of different measures
- to review whether some evaluation or design omissions or mistakes exists.

Independent review and identification of evaluation and design omissions or mistakes are very important because of extreme complexity of the seismic safety program. Practically this extent and depth of seismic re-evaluation and upgrading of a WWER had no precedents.
The seismic PSA was performed using methodologies described in the documents: IAEA-TECDOC-724, Probabilistic safety assessment for seismic events, IAEA, Vienna, October 1993 and EPRI TR 103959, Methodology for Developing Seismic Fragilities, EPRI, June 1994.
Some specific exceptions were made to the standard method of developing fragilities, e.g. simplified non-linear calculations of sliding, rocking of un-anchored equipment, and collapse of an un-reinforced masonry.
The risk model is primarily composed of equipment that is part of the Safe Shutdown Technology Concept as well as the equipment required for containment isolation and integrity. In addition, some equipment that are part of the normal electrical power supply and power conversion systems are included to take credit for the availability of these systems at lower levels of earthquake shaking and to assess the consequences of their failure as it may affect seismic safety relevant components.
As per experience the lower half of the fragility curve is the most important in the computing the unconditional probability of failure of the individual basic events. Therefore 100.000 years return period surface spectrum was selected (higher than the DBE) as the best representation of the surface response shape. Since the complete hazard is defined as PGA at rock, the relationship between the rock and surface uniform hazard response spectra at the 100.000 year return period have been included into the derivation of the structural response factor. Development of structural fragilities includes this transfer function into the structural response factor. Calculations using a latin-hypercube-simulation process were conducted to develop fragility curves for incipient liquefaction, gross liquefaction and ground settlements of 5, 10 and 20 cm. For components and structural elements that just meet the DBE requirements, the selected response spectra shape should provide the highest contribution of risk. As the seismic motion at rock increases the response above the soil is attenuated, thus the upper portion of the fragility curves are conservative. For very rugged components, the fragility derived is conservative since, the surface motion can likely never reach a level high enough to fail the component. A few cases were found where design errors and omissions resulted in very low capacities and the use of a 100.000 years return period surface spectrum is optimistic. The critical points of the main building complex were the steel superstructures

of the turbine building and the reactor hall. The reinforced concrete reactor block was judged seismically rugged.
Calculation of the strength factors of various structural elements along the seismic load paths in the longitudinal direction revealed that the weakest elements are the existing bolted connections of the vertical braced frames of the Main-Building-Complex steel superstructure. The bolted connections of the existing bracing members have the lowest strength factor. The critical bolted connections were identified. Failure of the bolted connections near the top of the superstructure might lead to failure of collectors. The panels cannot react tension loads transferred by the failures of collectors, thus, vertical splitting of structure could occur. The strength factors of the most critical elements range from 0.5 to 0.63. There is no ductility associated with this brittle failure mode. It is estimated that the composite strength factor of 0.6 represents the ultimate strength factor for the reactor building superstructure and 0.5 for the turbine building.
The total core damage frequency (CDF) was found about 3*10-4/a.

As a consequence of this (independent) probabilistic assessment of seismic safety different measures were identified, which were recognised as necessary to reach the required level of seismic safety, e.g. fixing of some masonry walls not reinforced before, fixing of some untested relays and cabinets, reinforcement of block walls in the Diesel generator building, etc.

The upgrade of the bolted connections in the reactor and longitudinal electrical gallery/turbine hall reduce alone the core damage frequency (CDF) by factor of 3.
Once these measures will be implemented, the liquefaction will be the dominant contributor to the CDF.
The CDF after completion of the measures mentioned above will be is equal to 3*10-5/a.
Today all the measures identified as necessary by the Seismic PSA are implemented except of improving the joints in the reactor and turbine hall steel-frames. The implementation of these upgrading measures is going on recently. Consequently the contribution of the seismic events to the CDF will be the same as for other major contributors.

Seq. No	Country	Article	Ref. in National Report
80	Slovenia	Article 14	section 3.7.6, p 48

Question/ Comment: Seismic contribution to CDF is of an order of magnitude larger than the contributions by internal events, internal fire and flooding in both operation and shutdown modes.
Has Paks NPP considered modifications to lower the seismic contribution?

Answer: The seismic PSA studies have been started in the final phase of the seismic upgrading program. The seismic PSA performed for the plant before essential seismic upgrading implemented would identify large number of trivial contributors of risk. The objectives of the seismic PSA was
• to quantify the safety of the plant in case of an earthquake
• to confirm the procedure of cool-down and heat removal
• to judge about contribution to the safety of different measures
• to review whether some evaluation or design omissions or mistakes exists.
Independent review and identification of evaluation and design omissions or mistakes are very important because of extreme complexity of the seismic safety program. Practically this extent and depth of seismic re-evaluation and upgrading of a WWER had no precedents.
The seismic PSA was performed using methodologies described in the documents: IAEA-TECDOC-724, Probabilistic safety assessment for seismic events, IAEA, Vienna, October 1993 and EPRI TR 103959, Methodology for Developing Seismic Fragilities, EPRI, June

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Some specific exceptions were made to the standard method of developing fragilities, e.g. simplified non-linear calculations of sliding, rocking of un-anchored equipment, and collapse of an un-reinforced masonry.
The risk model is primarily composed of equipment that is part of the Safe Shutdown Technology Concept as well as the equipment required for containment isolation and integrity. In addition, some equipment that are part of the normal electrical power supply and power conversion systems are included to take credit for the availability of these systems at lower levels of earthquake shaking and to assess the consequences of their failure as it may affect seismic safety relevant components.
As per experience the lower half of the fragility curve is the most important in the computing the unconditional probability of failure of the individual basic events. Therefore 100.000 years return period surface spectrum was selected (higher than the DBE) as the best representation of the surface response shape. Since the complete hazard is defined as PGA at rock, the relationship between the rock and surface uniform hazard response spectra at the 100.000 year return period have been included into the derivation of the structural response factor. Development of structural fragilities includes this transfer function into the structural response factor. Calculations using a latin-hypercube-simulation process were conducted to develop fragility curves for incipient liquefaction, gross liquefaction and ground settlements of 5, 10 and 20 cm. For components and structural elements that just meet the DBE requirements, the selected response spectra shape should provide the highest contribution of risk. As the seismic motion at rock increases the response above the soil is attenuated, thus the upper portion of the fragility curves are conservative. For very rugged components, the fragility derived is conservative since, the surface motion can likely never reach a level high enough to fail the component. A few cases were found where design errors and omissions resulted in very low capacities and the use of a 100.000 years return period surface spectrum is optimistic. The critical points of the main building complex were the steel superstructures of the turbine building and the reactor hall. The reinforced concrete reactor block was judged seismically rugged.
Calculation of the strength factors of various structural elements along the seismic load paths in the longitudinal direction revealed that the weakest elements are the existing bolted connections of the vertical braced frames of the Main-Building-Complex steel superstructure. The bolted connections of the existing bracing members have the lowest strength factor. The critical bolted connections were identified. Failure of the bolted connections near the top of the superstructure might lead to failure of collectors. The panels cannot react tension loads transferred by the failures of collectors, thus, vertical splitting of structure could occur. The strength factors of the most critical elements range from 0.5 to 0.63. There is no ductility associated with this brittle failure mode. It is estimated that the composite strength factor of 0.6 represents the ultimate strength factor for the reactor building superstructure and 0.5 for the turbine building.
The total core damage frequency (CDF) was found about 3*10-4/a.

As a consequence of this (independent) probabilistic assessment of seismic safety different measures were identified, which were recognised as necessary to reach the required level of seismic safety, e.g. fixing of some masonry walls not reinforced before, fixing of some untested relays and cabinets, reinforcement of block walls in the Diesel generator building, etc.

The upgrade of the bolted connections in the reactor and longitudinal electrical gallery/turbine hall reduce alone the core damage frequency (CDF) by factor of 3.

Once these measures will be implemented, the liquefaction will be the dominant contributor to the CDF.
The CDF after completion of the measures mentioned above will be is equal to 3*10-5/a.
Today all the measures identified as necessary by the Seismic PSA are implemented except of improving the joints in the reactor and turbine hall steel-frames. The implementation of these upgrading measures is going on recently. Consequently the contribution of the seismic events to the CDF will be the same as for other major contributors.

After implementation of the measures identified as necessary on the basis of seismic PSA the contribution of seismic events to the CDF will be the same order of magnitude as of the other contributors.
Most of the upgrading measures identified on the basis of seismic PSA are already implemented. The improvement of the joints is going on.

Seq. No	Country	Article	Ref. in National Report
81	Germany	Article 14.1	p. 47, 3.7.6

Question/ Comment: Were analyses performed regarding the reliable long-term operation of the HP ECCS pumps in the recirculation mode after LOCA (blocking of the pumps by particle debris)?

Answer: The problem of containment sump blockage (by debris) after a LOCA event in Paks is solved by a special filter system. The latest investigations in this area are in progress.

Seq. No	Country	Article	Ref. in National Report
82	Germany	Article 14.1	p. 47, 3.7.6

Question/ Comment: According to EOP, before starting to cool down the plant, the shutdown boron concentration has to be established. For some BDBA sequences a faster start of the plant cooldown would be preferable. Does the Severe Accident Management Guidance (SAMG) consider that?

Answer: At this moment only the strategic plan related to handling of severe accident situations has been completed. The Severe Accident Management Guidance is expected to be issued at the end of 2007. They will contain the procedures related to a fast cooling down parallel with partial boration.. Actually there is a chapter in the symptom oriented EOP procedures, which contains the step and conditions of urgent cooling down parallel with partial boration.

Seq. No	Country	Article	Ref. in National Report
83	United States of America	Article 14.1	Section 4.3.1

Question/ Comment: PSA analyses are mentioned in Section 4.3.1. Were these analyses performed by the operating organization? Are there any plans for HAEA to develop independent assessment tools?

Answer: The PSA analyses were performed by the operational organisation with the technical support of several research institutions. All the PSA models of the Paks NPP are available for the Regulatory Body to perform analyses when it becomes necessary. Beyond the PSA models there are three specific PSA tools at the HAEA, which were developed for regulatory use by the HAEA's Technical Support Organisation.

The Risk Supervisor System (RSS) was developed to analyse the effect of certain operational interactions on the core damage frequency as a function of time. The scope of interactions covers equipment/train switch-overs from operational to stand-by mode as well as unexpected outages. The planned risk profile of an annual campaign period as well as the actual follow-up risk profile can also be calculated during the plant safety supervision. The RSS system has been applied for all the four units of the plant.

The Precursor Event Analysis System (PEAS) was designed to calculate the conditional core damage probability considering the conditions of a certain unexpected event (component failure or initiating event). The PEAS system can be operated in an interactive way on a case by case basis.

The Core Damage Risk Prediction System (CDRP) was developed to support decision-making in a nuclear emergency situation at the Paks NPP and it is installed at the HAEA's Centre for Emergency Response Training and Analysis (CERTA). The CDRP helps HAEA's Emergency Response Staff to make very fast prognoses on the possible accident sequences in a probabilistic approach. The CDRP system can be easily operated in an interactive way on a case by case basis.

Seq. No	Country	Article	Ref. in National Report
84	United States of America	Article 14.1	Section 3.7.6

Question/ Comment: Did the decrease in core damage frequency to half its original value result from improvements in the PSA modeling technique or from modifications made to the units? What improvements were most effective in reducing the CDF?

Seq. No	Country	Article	Ref. in National Report
88	Japan	Article 14.2	P.47/L.37

Question/ Comment: Thanks to the implemented measures the safety of the units has increased. The core damage frequency has been decreased to half of the original value for both the operating rector and the reactor shutdown for refueling or maintenance

It is reported the core damage frequency has been decreased to half of the original value and the annual average probabilities of core damage were for the four units in sequence: 3.8x10-5; 3.3x10-5; 4.4x10-5; 3.6x10-5.
What are the major contributors to the reduction of core damage probabilities?

Answer: The existing PSA studies are updated annually. All the safety related plant modifications and changes in the reliability characteristics of plant equipment and/or plant personnel are modelled, the PSA results and documentation are updated as necessary. According to these assessments, the reported considerable risk reduction can be principally attributed to the safety upgrading measures (SUM) that have been implemented at Paks NPP up to now. In addition, there were also changes in the understanding of the plant, due to operational experience, data collection, and certain improvements in the modelling techniques.

The plant's program of safety upgrading measures was completed in 2002. Implementation of the program significantly improved the safety of the plant, and in the meanwhile provided an opportunity to utilise the operational experience accumulated up to now all over the world. By performing the safety upgrading measures, safety of the Paks NPP attained the safety level of western NPPs of similar age.

Efficiency of the SUM could be characterised either by the probability of core damages occurring during the 330-day fuel cycle from initial event of internal origin, fire, internal flooding, or by that of events during the outage for refuelling following the cycle. This probability value today is about $5{,}0 \cdot 10^{-5}$/ry indicating an improvement of more than one order of magnitude during last years, mainly resulted by the implemented safety upgrading measures.

The AGNES project aiming to a comprehensive evaluation of the safety in the mid-nineties

– on the ground of deterministic and probabilistic (PSA) analyses – quantified the safety level of the Paks NPP. The expected annual average core damage frequency, CDF referring to the nominal output, internal technological initial event in that time amounted about $5{,}0\cdot10^{-4}$/ry. The AGNES project provided the ground for the recently completed SUM program, as it finalised the list of safety upgrading measures, and defined their priorities. By PSA assessments became possible to specify those measures that have outstanding contribution to the core damage risks, thus distribution of risk factors could be equalised. The SUM list became part also of Authority tasks arising from the Periodic Safety Review:

- The most important SUM was the modification of the auxiliary emergency feedwater system. Having the system relocated into the reactor hall, the nominal, internal CDF value diminished by one order of magnitude per units.
- The further most important measures were the elimination of forced loss off-site power signal and the realisation of emergency cooling by primary circuit feed-and-bleed. By them, the above mentioned risk has been reduced to its half or one-third part.
- The project aiming to increase the reliability of human activities was also important. Its effect could be hardly expressed by numbers, its impact however on safe and reliable work performance is significant.
- Refurbishment of the reactor protection system has an essential effect in improving the operational safety, as a protection system on the world level has been installed on the units.
- Protection of containment sump against clogging with redesigning of the sump strainers;
- Prevention of the refilling of the tanks of the low-pressure emergency core cooling system after they have been emptied;

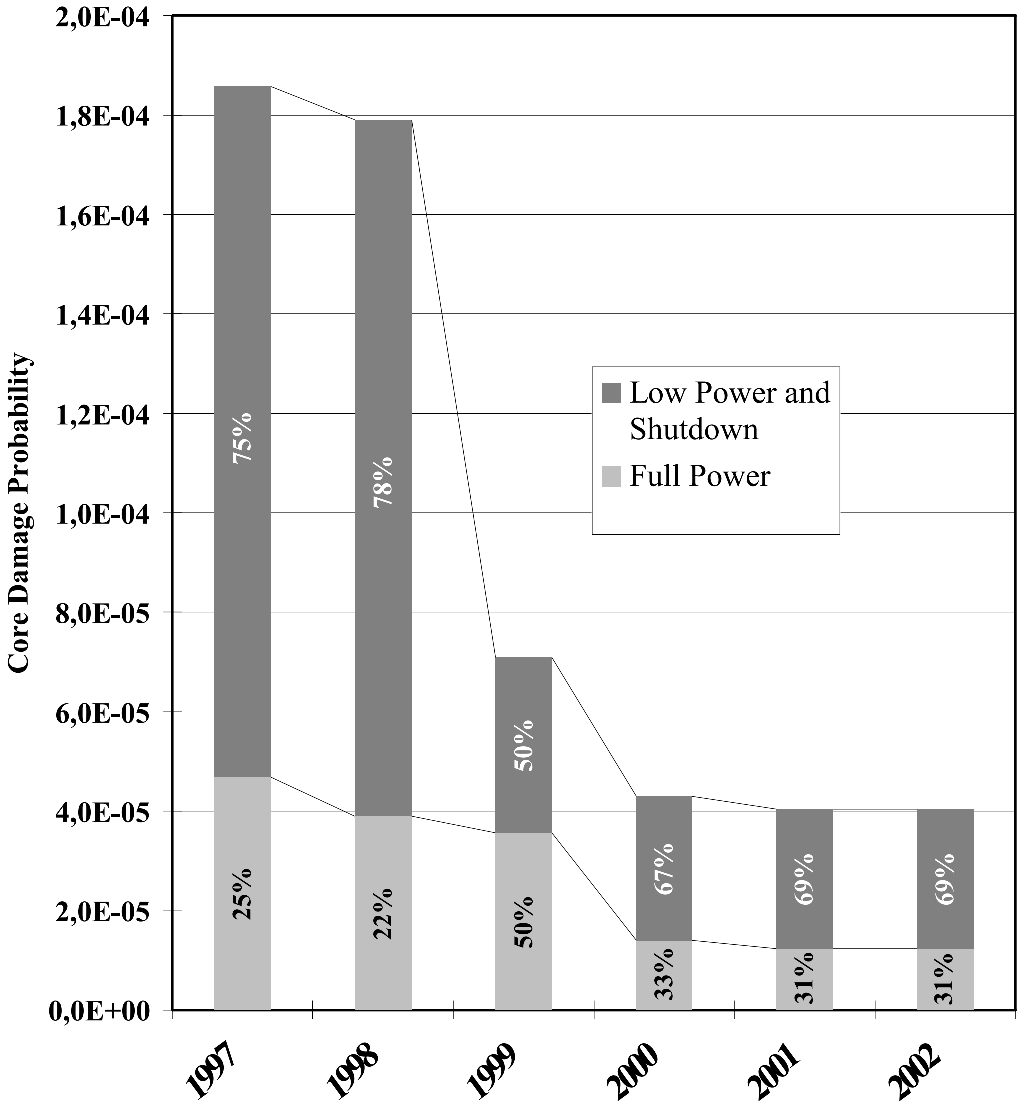
PSA analyses referring to non-nominal output were valid first for the 1995 reference time. One can see unambiguously from the figure that the risk of core damage during the outage following the cycle for refuelling is significant compared to the nominal state. The effect of the safety upgrading in this state, since 1999, is outstanding. In this period, the measures were implemented in the following areas:

- Safety revision of the heavy load crane operations in the reactor hall, definition of the routing for safe crane movements.
- Measures against erroneous routing in operating states of open reactor vessel, reduction of risk of occurrence for emergencies beginning with loss of coolant in the secondary circuit.
- To reduce the failures remaining from maintenance works, proper design of the tests, utilisation of data from experience.

Risk analyses for fire and flooding were completed first in 1999. Implementation of upgrading measures originating from them since 2000 resulted in further safety improvement. Some more important among them:

- Updating of the fire signalisation and extinguishing control.
- Development of fire extinguishing system for the main circulation pump (MCP) motors.
- Refurbishment of the turbine foam fire extinguishing system.
- Installation of emergency oil and hydrogen discharge system for the generators.

Through the Living PSA program the core damage risk has been systematically followed during recent years. The figure enclosed illustrate its trend demonstrating the favourable effects of SUMs described above.



Seq. No	Country	Article	Ref. in National Report
85	France	Article 14.2	§3.7.4 - p. 46
Question/ Comment	Could Hungary indicate if any measures are intended to be taken to understand and eliminate the false signals that are triggering the earthquake and alarm protection?		
Answer	The false signals, i.e. the spurious scrams are totally excluded in the implemented system at the Paks NPP. The acceleration trigger-crossing will activate the closing of some valves, which separate the fixed and non-fixed parts of systems from each other. This separation does not affect the operability of the plant. The operator actions after an earthquake are based on the CAV OBE-exceedance criterion, as described below: A seismic instrumentation has been installed on each unit of the plant in 1993. The instrumentation consist of seismic switches mounted on the base mat, sensitive		

accelerometers registering the response at the characteristic points of the structure, appropriate data collection system and voting logic. Two free field stations are installed at the plant too.
The hardware would allow adaptation of different principles of shut-down of the plant in case of an earthquake.
In 1993 it was also recognised that the probability of spurious scram due to ambient vibrations or impacts or failure of instrumentation is too high, consequently the automatic scram based on the acceleration trigger level crossing is disadvantageous and could not be justified by safety reasons.
After extensive study of international practice the cumulative absolute velocity (CAV) criteria of continuous safe operation (and for the cases of exceeding OBE) had been adapted. The necessary data acquisition and evaluation procedures and software have been developed. An emergency procedure exists which determines the operator activity after the earthquake. A comprehensive guidance was elaborated for assessing the post earthquake situation at the plant. The applicability of CAV criteria for the continuous safe operation is based on results of capacity evaluation of the SSC as well as the experience data behind of the response spectrum and cumulative absolute velocity limits.
It means, that no automatic reactor scram will be activated, if an earthquake occurs.
There is a specific condition valid for Paks NPP to be considered. According to the Hungarian regulation, reactor cool down and heat removal unlimited in time have to be ensured after an earthquake. These requirements implicitly mean that essential parts of the plant have to be operable after an earthquake and that those part which are not needed for the heat removal (and not fixed) have to be separated from the operable and fixed ones by quick-closing valves. This separation will happen automatically based on the (OBE) acceleration trigger level crossing. However this separation itself does not disturb the continuation of normal operation. Therefore these conditions might be kept, while the signal records are evaluated, the CAV criteria and those of exceeding of OBE are defined.

Seq. No	Country	Article	Ref. in National Report
86	France	Article 14.2	§3.7.6 - p. 47
Question/ Comment	Could Hungary explain on which bases reconstruction of the containment sumps was carried out? Was there a co-operation with members of the Club of Operators of WWER-400?		
Answer	After a false safety valve operation at the Barsebäck NPP in Sweden the containment sump filled up with isolating material and thus the coolant flow to the sprinkler and ECCS pumps was blocked. After this event the countries having nuclear power plants studied the problem in the frame of safety enhancing measures. Paks NPP contracted with the Finnish IVO International Ltd. to analyse the system at Paks and to offer solutions to decrease the risk of containment sump blockage. Finally from three versions of modification the reconstruction of the containment sump was chosen with the aim of increasing its filtering capacity. IVO International Ltd. and the Hungarian ETV ERÕTERV Rt. Have made the detailed design of the modification. As the result of the reconstruction the filtering surface was increased more than four times its original value, consequently the risk of the blockage of the coolant flow became negligible. The geometry of the filter elements is such that if still the filtering capacity decreased in a large extent then stopping of the pumps for a short time would cause a so-called self-correction of filtering. In this case most of the isolating materials, blocking the flow, would come off the filtering surfaces automatically.		

Seq. No	Country	Article	Ref. in National Report

87	Germany	Article 14.2	p. 46, 3.7.5
Question/ Comment	The next PSRs have to be completed before 2008 and 2010 respectively. Does HAEA plan to update the requirements on PSRs, taking into consideration the updated IAEA guide and the experience feedback from the already performed PSRs?		
Answer	Requirements to be put forward in 2008 and 2010 are included into the Nuclear Safety Regulations entering into force in the nearest future. The requirements have been formulated by taking into consideration the prescriptions in NS-G-2.10. The HAEA NSD wishes to make use of the previous PSRs. The main lessons learned from the previous PSRs that can be utilised during the next ones are: • the purpose of the FSAR actualisation and of the PSR is to be clearly defined and the scope, method and requirements of the PSR shall be defined in harmony with the goals, • because of the decentralised nature of the Hungarian authority system, all concerned authorities shall be involved into the process, • the review shall not solely mean the shear review of the documentation submitted to the authority, but the authority needs to take part in certain examinations performed by the operator		

Seq. No	Country	Article	Ref. in National Report
89	Croatia	Article 15	Ch.3.8.6, p.56
Question/ Comment	It is stated that HAEA signed an agreement with authorities responsible for radiation safety to «harmonize the inspections, the investigations of technical radiation protection and radiation protection, and to ensure full scope supervision of these fields within the nuclear installations» Could you provide more information about this agreement and experience in such formalized cooperation with those authorities?		
Answer	As described in the National Report in section 3.8.6., Hungary has a distributed regulatory system. The responsibility for general radiation protection is shared among three authorities. Additionally, as a result of the Hungarian rules, which do not explicitly define borderlines, to avoid overlaps or deficiencies the mentioned agreement was signed by the Chief Medical Officer of the State Public Health and Medical Officer's Service (SPHAMOS) and the Director General of the Hungarian Atomic Energy Authority (HAEA). In the framework of the agreement, the parties agreed that if the subject falls into the responsibilities of both authorities, they hold meetings before they conclude their license review process. Parties initiate expert meetings, in order to ensure the achievement of objectives of this agreement and fulfil their co-authority responsibilities. The agreement also states that various organizational units of each party inform each other on actions of mutual interest and that they provide each other with their annual report. The organizational units let each other have access to their databases and hold joint inspections for initiatives of any party. In the framework of this agreement, the HAEA NSD has fruitful cooperation with various organizational units of SPHAMOS's. Examples to that are also given in the National Report, such as after each inspection implemented by representatives either of the SPHAMOS or of the HAEA NSD inspection records are sent to the other authority. Besides, after each inspection resident inspectors of the authorities hold a consultation and inform each other on their activities conducted and experience gained. In addition, several joint inspections have been held following the agreement was signed.		

Seq. No	Country	Article	Ref. in National Report
90	Korea, Republic of	Article 15	

Question/ Comment	In relation to Table 3.8.2-1 Dose Limits, the remark 2) of the Table states that pregnant women are not permitted to be occupationally exposed. However, the ICRP 60 recommendations do not prohibit pregnant women from being occupationally exposed, but recommend that during the remainder of the pregnancy the equivalent dose to the surface of the woman's abdomen should not exceed 2 mSv (1/20 ALI in case of internal exposure). Please explain the rationale behind this regulation.		
Answer	The prohibition of pregnant women from being occupationally exposed was introduced into Hungarian legislation far before the publication of ICRP 60. Hungarian Authorities did not intend to relax protection of pregnant and have been keeping the traditional, more rigorous, formulation.		
Seq. No 91	Country Austria	Article Article 16	Ref. in National Report
Question/ Comment	The Report states that the completed new nuclear emergency response plans will be tested in 2004. What are the preliminary results of the testing in 2004?		
Answer	The tests of the new emergency response plans were performed through a number of drills, minor and complex exercises. According to the results the main achievement of the new plans is that the roles and responsibilities among the national organizations are well established and clarified. It is also observed that the plans enhance the efficiency and effectiveness of the emergency response of the various organizations as well as of the entire national system. The main improvements were identified as follows: - better and more accurate communication between on-site and off-site organizations, - quicker classification of events and notification of the off-site organizations, - better assessment and understanding of the emergency situation. The testing, however, also drew the attention to certain weak points: - a better utilization of expertise is necessary (sometimes delegating members from departmental or regional organizations into national boards exhausts the available expertise of the given organization), - some procedures shall be more precisely defined (converging to single decision from different decision-support recommendations; feed-back to the participating organizations on the implemented response actions), Based on the comprehensive evaluation of the results the working group for improving the national emergency response plan will be re-convoked to incorporate the necessary modifications into the national plan.		
Seq. No 92	Country Austria	Article Article 16	Ref. in National Report
Question/ Comment	Is the off site emergency planning in Hungary considering insights of probabilistic evaluation of accidents sequences and expected release categories? Are you prepared to share those information with your neighboring states in order to enable optimization of their (i.e. neighbor's) emergency preparedness for nuclear accidents?		
Answer	The level 1 and 2 PSA are already available for the Paks NPP, although currently a reviewing process is in progress. One of the main finding of the level 1 assessment is that the contribution of the events possibly occurring in shutdown state to the CDF is more significant than it was preliminary expected. This initiated to turn the attention to the assessment of that type of emergencies, e.g. calculating scenarios and source-terms for that type of situations as well. For that purpose a longer term research project was launched with the participation of a TSO organization.		

As a result of the level 2 PSA a set of predefined source-term groups were also elaborated. Based on that, within the frame of a European project, a code is under development (SPRINT), which will utilize these results for in-accident evaluation and also for training matters. From Hungarian side a TSO organization participates in the project, which provided the HAEA Emergency Response Organization with the first version of the software for testing.
Sharing of the asked information, after the regulatory approval of the PSA assessment, is certainly possible after mutual agreement of the involved countries.

Seq. No 93	Country Austria	Article Article 16	Ref. in National Report

Question/ Comment	Apart from the notification of an accident as required by the Convention on Early notification will the Hungarian emergency authorities and/or NPPs in Hungary be able to provide estimates of expected source term before the release (i.e. during an accident, when a release becomes imminent) as well as actual source term and the local weather data at the time of release?
Answer	The HAEA Emergency Response Organization (ERO), as national responsible organization for the international communication (international contact point and competent authority) during nuclear emergency situation, according to the common practice in wider-scope emergency response exercises, provides information on the evolved situation as required by the convention (i.e. EMERCON forms by fax, ENAC site) to the IAEA. Additionally, according to the bilateral agreements with neighbouring countries the HAEA ERO provides the same information directly to eight states.

The HAEA ERO has the legal right and the logistic tools for gathering the necessary information and to make diagnosis and prognosis in assessing the nuclear and radiological conditions.

In case of a nuclear emergency the international notification may be performed within a time interval that includes the time necessary for the notification from the plant, the notification and setting up of the HAEA Emergency Response Organization and the filling-in of the forms. This interval depends on the type of incident or accident; thus except for some special cases (e.g. so-called fast breaking events) the analysis is scheduled to be performed before the release. Beyond the notification the HAEA regularly performs further information on the event within the frame of the appropriate forms required by the Convention.

Based on these arrangements it can be stated that the HAEA ERO is prepared to perform due time estimation of the source term for a later release and also for an actual release, as well as the local meteorological data at the time of the release. Accordingly the required information is available at the HAEA ERO.

Seq. No 94	Country Canada	Article Article 16	Ref. in National Report 3.9.4, page 61

Question/ Comment	On top of page 61, subsection 3.9.4 of the report describes an “on-line, real time computer code” used at NPP to support assessment of radiological conditions during an emergency. Please describe how effective was the performance of this computer tool during the 10 April 2003 incident at Paks.
Answer	The computer code (named BALDOS) was not used right after the incident, because the severity of the incident was not recognized at the time of its occurrence. The day after the

incident the emergency response team was partially activated and it started to use BALDOS to check if the release measured by the Dosimetry Information System of the plant may entail any off-site consequence. The result was negative.
The substantial damage to fuel assemblies was revealed later (when the lid of the cleaning vessel was removed) and then the emergency team was activated again. This time the situation was classified as „alert” (according to IAEA emergency classification) and it was officially declared. At that time the release already decreased to a level where there was no need for evaluating the on-line radiological situation since numerous environmental measurement data were available. Nevertheless, some calculations were performed again, but the result was the same, no off-site consequences must have been considered.
After the event a program for cross-checking the different decision-support codes was initiated with the participation of the NPP and other expert organizations. The results of the BALDOS, calculated for the vicinity of the NPP, showed good agreement with the results of the other codes and with the measurement results.

Seq. No	Country	Article	Ref. in National Report
95	Croatia	Article 16	Ch. 3.92, p. 59

Question/ Comment: Could you explain the connection between regular radiological monitoring system and early warning system in the case of nuclear emergency? Which neighboring countries do you exchange radiological data with and in what way?

Answer: In Hungary there is an early warning and a regular radiological monitoring system. The early warning system is an on-line monitoring system performing ten-minute measurements and sending information to a data centre operated by the Directorate General for National Emergency Management (DGNEM). This system consists of 75 stations (additional 9 are under installation) located all over the country. In case of a nuclear emergency the data centre (its duty officer) receives automatic warning from the system, and the affected stations automatically switch to emergency mode (with a more frequent measurement rate). Upon the warning of the system, if necessary, the Hungarian Nuclear Emergency Response System may be activated.
After the notification, the responsible organization of the system (on the basis of the Ministry of Health) orders for switching the regular radiological monitoring system into emergency mode. This system involves the 75 stations of the early notification system and also a number of other measurements of different departmental and regional organizations (fixed and mobile laboratories, on-scene and field measurements, sampling, etc.).

Exchanging of radiological data is the responsibility of the international data exchange centre operated also by the DGNEM. Currently the situation is as follows:
- the centre regularly sends EURDEP data to the EURDEP centre to Ispra, Italy and to Austria;
- the centre exchanges radiological data with Slovakia through the Hungarian National Meteorological Service;
- EURDEP data is regularly received from Croatia and regular sending of EURDEP data to Croatia is under introduction,
- Hungary receives EURDEP data from Slovenia on a daily basis, the data exchange will be introduced soon.

Seq. No	Country	Article	Ref. in National Report
96	Romania	Article 16	page 57

Question/ Comment: Concerning the April 2003 incident, could you comment on the local contamination induced by the event on the capacity of the operational monitoring system?

Answer According to the measurements of the operational monitoring system, the following values characterize the released airborne radioactivity:

Period	Noble gases [TBq]	Iodine (^{131}I) equivalent [GBq]
10th April	12	143
11th April	160	204
12th April	25	7.0

The monitoring stations located within the 1.5 km vicinity of the plant (consisting of 9 stations measuring gamma-dose-rates) had not shown any increase above the daily fluctuations around the circa 100 nSv/h average value, except for one station showing a marked increase up to 260 nSv/h, for a short time on 11th April.

Seq. No	Country	Article	Ref. in National Report
97	Slovakia	Article 16	3.9

Question/ Comment Please explain who and how determines the emergency planning zone around the nuclear facilities. What is the basis for the zone specification? Which criteria do you apply to specify the zone area? Do you use/accept any probabilistic arguments to determine the zone area?

Answer The Emergency Planning Zones around the nuclear facilities in Hungary are defined by the National Nuclear Emergency Response Plan. Currently this plan is not a law; however its status is mandatory concerning nuclear emergency preparedness matters. Its force coming from the decision issued by the Governmental Coordination Committee established to preparing for emergency situations and acting as decision-making body in case of emergencies.

The specification of the zones is principally in concert with the recommendations by the IAEA (TECDOC-953). This document suggest intervals for the radii of the zones and gives guidance on what to consider (intervention levels, practical issues) for defining the boundaries. The developers of the national plan prepared dispersion calculations for scenarios that may be postulated for the given facilities in order to determine the adequate zone radii (taking into account the intervention levels determined by law) and also considered the local conditions (towns, villages, transportation, economics) when the almost circular boundaries have been determined.

So far probabilistic tools (i.e. level 3 PSA) were not used to determine the area of the emergency planning zones and no need for using them was yet arosen.

Seq. No	Country	Article	Ref. in National Report
98	France	Article 16.1	§3.9.3 - p. 59

Question/ Comment Could Hungary indicate whether iodine prophylaxis is part of preventive measures for mitigating the consequences of radiological accidents and, if yes, could details be provided on the distribution system?

Answer Iodine prophylaxis is considered as preventive measure in Hungary. Iodine tablets are deposited in either pharmacies or mayor’s offices in settlements within a 30 km radius area around the NPP. In case of emergency, a staff directed by the Ministry of Health and the

	Operative Staff of the Governmental Co-ordination Committee distributes the tablets.		
Seq. No 99	Country France	Article Article 16.1	Ref. in National Report §3.9.4 - p. 60
Question/ Comment	Could Hungary indicate whether specific actions are scheduled in case of a fast kinetic accident, i.e. an accident leading to radiation exposure of the public within a short period of time?		
Answer	In case of emergency an alarm system warns the people in settlements within a 30 km radius area around the NPP. First sirens give alarm signals. Then loudspeakers release immediate information, and people are in advance instructed to switch in the radio receivers, since the main radio station of Hungary starts immediately informing the public. The most probable immediate action is advice on sheltering. This may be followed by stable iodine distribution and evacuation.		
Seq. No 100	Country United States of America	Article Article 16.1	Ref. in National Report Section 4.2.2, Secti
Question/ Comment	Section 3.9.4 states that the INEX 2000 international exercise identified remaining tasks and weak points in the emergency plan. Please discuss the insights gained and corrective actions taken as a result of this and other emergency exercises. Is there a schedule for developing protective action guidelines as part of a severe accident management strategy?		
Answer	At the time of the INEX-2000 exercise Hungary had a somewhat obsolete National Emergency Response Plan, although a working group for its revision had already been formed, therefore the exercise was a good opportunity to reveal the weak points. The exercise primarily raised problems with the responsibilities, roles and communication within the Hungarian Nuclear Emergency Response System (HNERS). The decision-making process was rather difficult and in certain steps poorly regulated, however the participating organizations acted and performed their activities as they thought to be done. Main areas where deficiencies were experienced: - Notification and activation of the HNERS in case of accident in foreign country. - Processing of meteorological data from the Hungarian National Meteorological Service to other HNERS organizations. - A number of errors and non-appropriate text-field in the forms elaborated for emergency communication. - Missed answer to international expert question due to discrepancy in the documentation system of the contact point. - Issues raised with positions within certain organizations concerning the lack of competence or over-redundancy. - Deviations from written procedures. Some other less important findings were also identified. A detailed plan of improving measures for handling these issues was approved and implemented. The items of this action plan were mainly handled in the frame of the National Emergency Response Plan which was finalized at the end of 2003 and in the connecting facility, departmental and regional level emergency plans. Based on the experience gained from the comprehensive exercise held in November, 2004 and from other smaller exercises the emergency preparedness was significantly improved on every level The tests of the new emergency response plans were performed through a number of drills, minor and complex exercises. According to the results the main achievement of the new plans is that the roles and responsibilities among the national organizations are well established and clarified. It is also observed that the plans enhance the efficiency and		

effectiveness of the emergency response of the various organizations as well as of the entire national system. The main improvements were identified as follows:
- better and more accurate communication between on-site and off-site organizations,
- quicker classification of events and notification of the off-site organizations,
- better assessment and understanding of the emergency situation.

The testing, however, also drew the attention to certain weak points:
- a better utilization of expertise is necessary (sometimes delegating members from departmental or regional organizations into national boards exhausts the available expertise of the given organization),
- some procedures shall be more precisely defined (converging to single decision from different decision-support recommendations; feed-back to the participating organizations on the implemented response actions),

Based on the comprehensive evaluation of the results the working group for improving the national emergency response plan will be re-convoked to incorporate the necessary modifications into the national plan.

The elaboration of severe accident management strategy for the only NPP in Hungary is at an initial phase. So far the development of protective action guidelines is not scheduled as part of the strategy, however protective action guidance exists as part of the national and lower level emergency plans, which discuss the details of deciding and implementing the off-site protective actions.

Seq. No	Country	Article	Ref. in National Report
101	Austria	Article 17	

Question/ Comment: The report states that evaluation of the seismic reinforcement and the implementation of the prescribed reinforcements were fully completed by the end of 2002. Are there plans to implement further measures with the goal that Paks NPPs reach the IAEA INSAG CDF safety target of 1×10 (-4)/a? #

Answer: The seismic PSA studies have been started in the final phase of the seismic upgrading program. The seismic PSA performed for the plant before essential seismic upgrading implemented would identify large number of trivial contributors of risk. The objectives of the seismic PSA was
• to quantify the safety of the plant in case of an earthquake
• to confirm the procedure of cool-down and heat removal
• to judge about contribution to the safety of different measures
• to review whether some evaluation or design omissions or mistakes exists.
Independent review and identification of evaluation and design omissions or mistakes are very important because of extreme complexity of the seismic safety program. Practically this extent and depth of seismic re-evaluation and upgrading of a WWER had no precedents.
The seismic PSA was performed using methodologies described in the documents: IAEA-TECDOC-724, Probabilistic safety assessment for seismic events, IAEA, Vienna, October 1993 and EPRI TR 103959, Methodology for Developing Seismic Fragilities, EPRI, June 1994.
Some specific exceptions were made to the standard method of developing fragilities, e.g. simplified non-linear calculations of sliding, rocking of un-anchored equipment, and collapse of an un-reinforced masonry.
The risk model is primarily composed of equipment that is part of the Safe Shutdown

Technology Concept as well as the equipment required for containment isolation and integrity. In addition, some equipment that are part of the normal electrical power supply and power conversion systems are included to take credit for the availability of these systems at lower levels of earthquake shaking and to assess the consequences of their failure as it may affect seismic safety relevant components.
As per experience the lower half of the fragility curve is the most important in the computing the unconditional probability of failure of the individual basic events. Therefore 100.000 years return period surface spectrum was selected (higher than the DBE) as the best representation of the surface response shape. Since the complete hazard is defined as PGA at rock, the relationship between the rock and surface uniform hazard response spectra at the 100.000 year return period have been included into the derivation of the structural response factor. Development of structural fragilities includes this transfer function into the structural response factor. Calculations using a latin-hypercube-simulation process were conducted to develop fragility curves for incipient liquefaction, gross liquefaction and ground settlements of 5, 10 and 20 cm. For components and structural elements that just meet the DBE requirements, the selected response spectra shape should provide the highest contribution of risk. As the seismic motion at rock increases the response above the soil is attenuated, thus the upper portion of the fragility curves are conservative. For very rugged components, the fragility derived is conservative since, the surface motion can likely never reach a level high enough to fail the component. A few cases were found where design errors and omissions resulted in very low capacities and the use of a 100.000 years return period surface spectrum is optimistic. The critical points of the main building complex were the steel superstructures of the turbine building and the reactor hall. The reinforced concrete reactor block was judged seismically rugged. Calculation of the strength factors of various structural elements along the seismic load paths in the longitudinal direction revealed that the weakest elements are the existing bolted connections of the vertical braced frames of the Main-Building-Complex steel superstructure. The bolted connections of the existing bracing members have the lowest strength factor. The critical bolted connections were identified. Failure of the bolted connections near the top of the superstructure might lead to failure of collectors. The panels cannot react tension loads transferred by the failures of collectors, thus, vertical splitting of structure could occur. The strength factors of the most critical elements range from 0.5 to 0.63. There is no ductility associated with this brittle failure mode. It is estimated that the composite strength factor of 0.6 represents the ultimate strength factor for the reactor building superstructure and 0.5 for the turbine building.
The total core damage frequency (CDF) was found about 3*10-4/a.

As a consequence of this (independent) probabilistic assessment of seismic safety different measures were identified, which were recognised as necessary to reach the required level of seismic safety, e.g. fixing of some masonry walls not reinforced before, fixing of some untested relays and cabinets, reinforcement of block walls in the Diesel generator building, etc.

The upgrade of the bolted connections in the reactor and longitudinal electrical gallery/turbine hall reduce alone the core damage frequency (CDF) by factor of 3.
Once these measures will be implemented, the liquefaction will be the dominant contributor to the CDF.
The CDF after completion of the measures mentioned above will be is equal to 3*10-5/a.
Today all the measures identified as necessary by the Seismic PSA are implemented except of improving the joints in the reactor and turbine hall steel-frames. The implementation of

these upgrading measures is going on recently. Consequently the contribution of the seismic events to the CDF will be the same as for other major contributors.

Considering the recent status of the Paks NPP, the contribution of the seismic event to the CDF is below of the IAEA INSAG CDF safety target of 1×10-4/a and will be finally (by the end of implementation of additional measures identified on the basis of seismic PSA) in the order of magnitude of 10-5/a.

Seq. No	Country	Article	Ref. in National Report
102	Germany	Article 17.1	p. 67, 4.1.6

Question/ Comment: Why is the macroseismic intensity 6 (MSK) used as the design basis value even though earthquakes of the intensity 8 (MSK) can occur every 40 to 50 years in Hungary?

Answer: The NPP site was selected in sixties. The site selection and characterisation followed the practice and regulations valid at that time in the Soviet Union. During the site selection and characterisation the site seismicity was assessed on the basis of earthquakes catalogue and isoseismal map of Hungary. The earthquake catalogue contains historical records dating back to the 4th century. It was found that the site seismicity can be set equal to intensity MSK 6. For the sake of correctness and to clarify the assumptions made during the siting in sixties: Although the recurrence of earthquakes with intensity 8 is approximately 50 years in Hungary, the seismic activity in the Pannonian region can be characterized as moderate, with significant variations in different tectonic domains. In the catalogue there are no evidences for significant activity in the region of the site. The closest larger event of intensity 7-7.5 was recorded at the distance from the site more then 50 km.

Seq. No	Country	Article	Ref. in National Report
103	France	Article 17.3	§4.1.5 - p. 67

Question/ Comment: Beyond actions taken to preclude nuclear facilities to withstand external phenomena, does Hungary study the on-site effect of flood (for instance, possible leak path via galleries and ducts and then the possible degradations of train A and B equipment)?

Answer: The investigation of internal floods as well as of internal fires was part of the PSA analyses. (internal flood PSA, fire PSA). The results of the analyses were, that the three train of redundant safety systems (mechanical, electrical and instrumental) are protected against common cause failures possible cased by internal flooding. This means, that in case of such an event only one of the three trains may fail.

Seq. No	Country	Article	Ref. in National Report
104	Austria	Article 18	

Question/ Comment: Are there safety analyses for the NPPs and the interim storage facility against terrorist attacks? Which measures have been taken to minimize the risk of terrorist attacks against NPPs and the interim storage facility?

Answer: Question:
Are there safety analyses for the NPPs and the interim storage facility against terrorist attacks?

In November 2001 (some weeks after 9.11) the Director General of the Hungarian Atomic Energy Authority convened the experts of the nuclear business and requested them to elaborate a study on the terrorist threaten of the Hungarian nuclear installation and connected areas.
The expert group finished the study in March 2002, in which the threaten of the nuclear installations was reviewed both on legal and technical bases. The analyses covered all the

interested areas:
- threaten in general
- national preparedness
- potential targets and countermeasures
- Paks NPP
- Budapest Research Reactor
- Budapest Technical University Training Reactor
- Interim Storage of Spent Fuels at Paks
- Storage of radwaste

The study summarized that these was no urgent step, the national and local preparedness are sufficient.
Nevertheless the study required some additional analyses and/or corrective actions.

In 2004 nearly the same expert group, also by the initiation of the HAEA, reviewed the lessons-learned of the previous study and terrorist-related events happened in the meantime.
The conclusion reaffirmed the basic statements of the 1st study, that presently the Hungarian nuclear installations do not play a central role or target in the field of hypothetic nuclear terrorism.

Question:
Which measures have been taken to minimize the risk of terrorist attacks against NPPs and the interim storage facility?

Paks NPP
Upgrading measures have been accomplished:
- since 2002 the organisations, involved into the physical protection system, have to perform exercises regularly on the basis of a long-term plan
- the plans of the reconstruction and upgrading of the physical protection system have been ready and approved by the regulatory body.
The reconstruction started in 2004 and the upgrading started in 2005.
- The emergency preparedness and accident management systems of the NPP have been reviewed and modified in harmony with new country-wide system.
(The New Comprehensive Plan came into force in April 2003.)

Interim Storage of Spent Fuels
- Upgrading measures have been accomplished:
The physical protection system of the Interim Storage was separated from the system of Paks NPP and started its independent activity (including the activity of the staff of its own) in November 2004.
- The Interim Storage has implemented some preventive measures against blackmail, demonstration, or green movement-oriented intrusion.

Seq. No 105	Country Austria	Article Article 18	Ref. in National Report
Question/ Comment	One possible safety improvement measure is the installation of hydrogen re-combiners inside the confinement. What are the results of the investigation of HAEA and the Paks NPP concerning the performance of hydrogen re-combiners under severe accident conditions? What measures have been taken to ensure that confinement structure failure		

does not occur during severe accidents owing to hydrogen combustion phenomena?

Answer

In the framework of the Safety Enhancement Measures (SEM) decided on the basis of the thorough safety analysis program called AGNES project 16 catalytic Hydrogen re-combiners were installed in each of the four containments of the Paks NPP. These re-combiners are only suitable to handle the expected Hydrogen production during design basis accidents. In case of beyond design basis or severe accidents, the Hydrogen production may be much higher, thus the existing re-combiners are not necessarily able to cope with it. This issue had been covered in the level 2 PSA project, which was completed in 2004. According to the analyses of this project, it is sufficient to install additional 30 such re-combiners to make sure that the Hydrogen concentration remains below dangerous levels under any circumstances. The details of these analyses are not available so far to the Authority, therefore the proposal is not yet approved. We also note that there is an ongoing international project (UK-Germany-Finland-Hungary) under PHARE financing, with the aim of determining the possible detailed Hydrogen distributions in case of severe accidents, by using sate-of-the-art CFD methods. The results from this project are expected by the end of 2005.

Detailed Severe Accident Management Guideline for the Paks NPP is under preparation. The detailed schedule of its elaboration and implementation is to be completed and agreed with the HAEA this year. The details of the planned SAMG are extensively based on the level 2 PSA studies and the general strategies are already considered and presented to the HAEA. These strategies include the application of active igniters and the use of filtered venting for limiting the containment internal pressure.

Seq. No	Country	Article	Ref. in National Report
106	Finland	Article 18.1	chapter 4.2.2

Question/ Comment

In the paragraphs 4.2.2 and 4.3.1 severe accidents are discussed. Are there plans to provide means for severe accident management, by backfitting the plant design or developing operational procedures? If yes, what are the main improvements.

Answer

As mentioned in paragraph 4.3.1 the previous Periodic Safety Review included some results of deterministic accident analyses. After this PSR elaborating of the level 2 PSA and the strategy of accident management procedures were prescribed. The level 2 PSA had been completed by the end of 2003 and submitted to HAEA NSD. After that a PHARE project has been initiated to evaluate the results and to establish further requirements. At the end of 2004 a strategy for the compilation of accident management procedures was submitted to HAEA NSD containing a set of tasks to be solved in the near or later future. Up to now the first version of the document has been discussed by the Paks NPP, HAEA NSD and AEKI (academic research institute, author of the strategy). After some corrections it has been agreed and a resolution has been issued that a detailed schedule of the realisation shall be ready by the end of 2005.

The most important tasks are

- At a short deadline a seismic study has to be done and its results shall be taken into account in the strategy.

- A symptom-based Severe Accident Guideline (SAG) is to be elaborated. This SAG will be available in printed form and on the process computers of the NPP and it will have the appropriate interfacing with the existing Guideline for DBAs.

- The existing Guideline for DBAs shall be completed for the shut-down states of the reactor and the events at the reactor pool.

- A thorough revision of the minimum set of measurements in accident situation shall be established.

- A list of technical measures: sizing of hydrogen recombiners, studying of the effectiveness of the Sprinkler system, flooding of the reactor shaft in case of the failure of the reactor vessel • ¨ strengthening of the maintenance door of the shaft, examination and - if necessary - modification of the electric power supplies, etc.

- Organisational changes.

Seq. No 107	Country United States of America	Article Article 18.1	Ref. in National Report Section 4.2.2
Question/ Comment	Section 4.2.2 states that Adue attention was not given to ... the basic design requirements related to protection against natural phenomena; ... external dynamic effects; and ... the unit control room@ when the power plant was constructed. How have these deficiencies been accounted for through backfitting or other means?		
Answer	Many analyses were carried out referring to natural phenomena using deterministic and probabilistic methods. As the result of these analyses it was found, that only the seismic events are above the screening criteria given by the regulation. According to this fact, the HAEA NSD issued an obligation to elaborate and carry out a seismic upgrading program. In this program, a great number of systems and structures have been reinforced (including the ceilings of the main control rooms). Additionally, a technological upgrading program has been completed, which assures the success of cooling down all four units at the same time after a design level earthquake (SL-2). The seismic upgrading program was completed in 31.12. 2002. In this moment there is no need for any further measure against the consequences of a natural phenomena.		

Seq. No 108	Country Germany	Article Article 18.3	Ref. in National Report p. 69, 4.2
Question/ Comment	Does the (improved) design consider ergonomic aspects and the man-machine interfaces?		
Answer	For the sake of safety, Act CXVI of 1996 on atomic energy, which is the highest level of regulations, stipulates that possibilities and limitations of the human performance shall be considered all along the lifecycle of the nuclear installations [4. § (5)]. This general principle is formulated as design requirements in Volume 3 of the Nuclear Safety Regulations (NSR) referred to at the beginning of section 4.2.1 in the Hungarian national report. Within Chapter 3 of the NSR (General requirements) a special part – 3.18, entitled „Human factor” – contains the requirements associated with the issues in the question. It consists of six paragraphs (3.159 – 3.165), the first requires that the ergonomic principles, the second that the man-machine interface shall be considered in the design. These very general statements are detailed in the remaining four paragraphs. Further requirements in connection with the human factor are contained in those parts which are related to the probabilistic safety assessment and design of the main control room. Since the actual NSRs entered into force (end of July, 1997) these requirements are implemented in practice. All plant modification applications of the Paks NPP (licensee) for approval are reviewed by the nuclear safety regulator taking all these requirements into		

	consideration. It is also worth mentioning that as a result of the periodic safety reviews of the NPP units, such improvement actions have been defined and implemented, which have relevance to ergonomic aspects and man-machine interface issues. The most remarkable among them is the introduction of symptom oriented operational procedures.

Seq. No	Country	Article	Ref. in National Report
109	Canada	Article 19	4.3.1, p 71 to 74

Question/ Comment	How will the scope and timing of the planned modifications to the Final Safety Analysis Report (FSAR) be influenced with the actual plant operating conditions after the 15 pending improvement measures would have been accomplished? What effect would the recent Paks incident have on the FSAR?
Answer	According to the regulations in effect modifications initiated by the operator shall be categorised according the IAEA NS-G-2.3 guideline. This categorisation determines whether the modification necessitates a permission from the authority. The authority performs a licensing procedure in cases of modifications belonging to category 1 and 2. During the licensing procedure the authority approves the changes in the respective chapters of the FSAR. The pages of the FSAR modified due to the performing of the approved technological modifications shall be included into the FSAR in a formal procedure. The FSAR mentioned in the National Report reflects the status of the plant at the end of 2002, thus from among the 15 safety increasing measures referenced in the question it contains those FSAR modifications, which are related to the technological modifications performed by the end of 2002. The rest shall be included into the FSAR during the next year. Only those changes are introduced into the FSAR, which may have longer term influence on safety or on systems related to safety. The version completed by the end of 2004 does not yet reflect any major change performed following the 2003 April incident. The related necessary FSAR changes shall first be described in conjunction with the application for license in principle for the recovery of damaged fuel elements. Assessment and approval of the proposed FSAR modifications shall be performed by the authority parallel to the evaluation of the license application and shall be include into the new version of the FSAR due next year. These modifications shall reflect those changes of the plant technology and circumstances that are long standing and result from the incident and/or the recovery, such as change in the composition of the liquid waste, increase of amount and change in the composition and distribution of the stored radioactive wastes, possible effects of changes on the decommissioning. Related to the incident in April 2003, the Paks NPP has performed a number interventions in order to improve its functioning, however these have no effect on the FSAR.

Seq. No	Country	Article	Ref. in National Report
110	Canada	Article 19	4.3.8, p 78 and 79

Question/ Comment	Please provide information on how trends are performed and monitored. What tools are used in these regards?
Answer	Paks NPP has been operating a safety performance indicator system based on IAEA TECDOC-1141 for collecting, gathering and illustating data of the various professional areas. The Safety Performance Indicators in order to reflect the plant safety performance on the basis of a wider range and specially arranged system of indicators. For every indicator goal values and thresholds for unacceptability were determined that helps to assess trends. Results of SPI assessment are presented in the Quarterly Report of SPI in the meeting of Safety and QA Management Committee. A regular managerial assessment of the values of

	safety indicator system has been introduced. The development of a web based computer program to support the assessment work is in progress.		
Seq. No 111	Country Canada	Article Article 19	Ref. in National Report Annex 7, page 117
Question/ Comment	Annex 7, page 117, of the report describes the short term tasks that the Authority undertook in follow up to the April 10 2003 Paks incident and identifies that the authority formulated those strict conditions with which the out-of-turn licensing procedure can be justified. Please identify those strict conditions.		
Answer	From the conclusions drawn from the events of April 01, 2003 HAEA NSD found it necessary and advisable to revise the conditions and rules of Expedited Review of urgent cases. NSD established the institution of Expedited Review for urgent license applications. The basic principles of Expedited Review are the following: • The currently effective legal means do not contain rules on Expedited Review; however, HAEA NSD accepts that such review may be necessary in certain cases. • The rules and conditions of Expedited Review process should be incorporated into a bilateral Agreement to define clear and mutually binding basic rules of cooperation. For governing Expedited Reviews, an agreement should replace an earlier memorandum. • Considering the limited resources available to HAEA NSD, both Paks NPP and HAEA NSD should be aware of the consequences of supporting an Expedited Review process by drawing resources away from concurrent other activities. • The Expedited Review process, as it is indicated by its name, provides accelerated timeline for review process, but otherwise is governed by the principles and rules of normal reviews. Expedited Review process may not be used to re-gain the time passed during preparation and filing, and it may not be used to re-classify safety priorities. • Expedited Review process may only be applied in cases supported by compelling evidence of the need for the accelerated process. HAEA NSD will weight the evidence provided in the application and will grant Expedited Review only when it is found necessary and appropriate. • Expedited Review may not lead to any decrease in the completeness or thoroughness neither in the submittal nor in the assessment. • HAEA NSD will consider the number and types of Expedited Review applications filed by Paks NPP, in calculating the overall safety indices qualifying the operations of Paks NPP." The Expedited Review execution is the next according to the Nuclear Safety Regulations: In appropriate and qualified matter, the Safety Director of Paks NPP may file a request for Expedited Review to the head of HAEA NSD. Such request may only be filed if the matter was not foreseeable and if its expedited review will materially contribute to the alleviation of an outstanding safety concern. The application must be filed in writing, and the Safety Director of Paks NPP is encouraged to follow up personally with the head of HAEA NSD. The head of HAEA NSD will grant or deny the request for Expedited Review. The head of HAEA NSD will consider the required and available resources, such as manpower, working hours, deadlines. The Director will consider the severity of safety concern, and decide if it is proportionate with the resources required and available for the Expedite Review. The decision will be based solely on these facts and factors, in each individual case. In addition to hardcopy files and archives, the HAEA NSD encourages the use of operative tools, such as telephone, fax, email, and personal consultation. The head of		

HAEA NSD will communicate his/her decision to the Safety Director of Paks NPP.

The Expedited Review affects only the timeline of the review, while all review principles remain unchanged. The applications must meet all the requirements regarding the contents and format.
HAEA NSD will not compromise nuclear safety in the consideration of requests for Expedited Review.

Seq. No 112	Country Canada	Article Article 19	Ref. in National Report Annex 7, page 118
Question/ Comment	In closing the detailed description of the Paks incident, the report states that “the upcoming years will inevitably bring several regulatory tasks requiring significant resources.” How would these resources be acquired? Please describe whether or not these resources would become permanent in nature since some of the recommended measures may become permanent when implemented.		
Answer	HAEA is financed partly from the state budget, partly from fees paid by the licensees, whereas HAEA is an entirely governmental organisation to which the general rules of civil service apply. Thus when attempting to acquire more resources then on one hand the licensee fees may be modified accordingly, on the other hand governmental approvals on increasing the budget and/or the number of staffs need to be reached. Both ways are tried by the HAEA management. In case extension of the resources as above is realised, it is meant to be permanent.		

Seq. No 113	Country Croatia	Article Article 19	Ref. in National Report Annex 7, p. 116
Question/ Comment	It is stated that Paks NPP elaborated and submitted a Comprehensive Action Plan to improve operating activities based on the experience gained from the incident in 2003. What is the status of this Action Plan and what concrete measures have been implemented for improving safety culture in operation of Paks NPP?		
Answer	The actions in the Comprehensive Action Plan (CAP) were initiated by the plant’s own investigation, by the investigation performed by the Hungarian Safety Authority and the IAEA review mission. The Action Plan is in the phase of implementation. This means that most of the short-term actions have already been completed. There are a number of actions with long-term effect particularly those dealing with safety management and safety culture improvement. This type of actions are in the implementation phase and some of them is planned to be completed by the year 2006. An IAEA mission is conducted at Paks in February 2005. The objective of the mission is to review the progress made by the plant since the OSART mission in 2001 and since the special review mission conducted in June 2003 after the Unit 2 incident (in fact to review the status of the implementation of CAP). The CAP contains the following specific measures for improving safety culture: The discussion of the experience gained during the managerial inspections performed in accordance with regulations included in the procedure ELJ-BIZT-05-05 has to become a regular item of the agenda at all levels of managerial meetings including the company management meetings. A task-oriented, systematic training system for employees of the safety organisation has to be elaborated. The objective of this system is to develop skills in conducting safety inspections, in management of safety issues, as well as appropriate interpretation of safety assessments and conditions.		

The Improvement Programme of the Management and Organisation System of the company accepted by the Management and the Board of Directors of Paks Nuclear Power Plant has to be implemented. This program has the following elements:

1. Vision and values – To review the corporate goal system
2. Optimisation of functioning of the organisation – For exact specification of responsibilities, to enhance decision making and to improve quality system and plant operation generally
3. Leadership improvement – To improve managerial knowledge of leaders
4. Human improvement – To improve communication and cooperation, attitude to quality
5. IT improvement – To support process review and plant control system

The company has to review, improve and rationalize the system of the company's decision-making mechanisms, as well as of the company forums and meetings. The objective of this activity is to provide that the place of decision making, as well as the person of decision maker and the relevant responsibility can be unambiguously defined, and that decisions are made at that level where the appropriate professional support is available, and that priority of the safety is provided during decision making.

The company has to review the methodology of safety culture surveys, including defining of actions potentially needed. The company has to perform these surveys regularly (e.g. in every two years). The scope of the surveys should cover both employees and managers of the company and should also include those of the strategic partners of the company.

The regular managerial assessment of the values of safety indicator system should be introduced
The company has to review the documents "Safety Policy" and "Quality Policy".

The company has to organize regular managerial forums at all levels of the organizational hierarchy. The human policy organisation has to elaborate a uniform consideration system for methodology of forums. The methodology has to contain constant "elements", for example safety, quality questions, as well as topical issues. The main items of these discussion topics should come from the top management, considering the most important questions and tasks of the company, but always emphasizing priority of the safety.

Seq. No	Country	Article	Ref. in National Report
114	Korea, Republic of	Article 19	Operations, p 82

Question/ Comment	In relation to paragraph 4.4, 'Plans concerning safety improvement', what are the major contents of the 'upgrading the safety culture' which was decided upon as one of the improvement measures after the serious incident on 10 April 2003 at unit 2'. Are there any indicators for assessing safety culture of nuclear power plant employees(individual level) ?
Answer	The Comprehensive Action Plan (CAP) defined after the incident contains the following specific measures for improving safety culture: 1. Evaluation of the experience gained during the managerial inspections performed in accordance with regulations included in the procedure ELJ-BIZT-05-05 has to become a regular item of the agenda at all levels of managerial meetings including the company

management meetings.

2. A task-oriented, systematic training system for employees of the safety organisation has to be elaborated. The objective of this system is to develop skills in conducting safety inspections, in management of safety issues, as well as appropriate interpretation of safety assessments and conditions.

3. The Improvement Programme of the Management and Organisation System of the company accepted by the Management and the Board of Directors of Paks Nuclear Power Plant has to be implemented. This program has the following elements:

• Vision and values – To review the corporate goal system
• Optimisation of functioning of the organisation – For exact specification of responsibilities, to enhance decision making and to improve quality system and plant operation generally
• Leadership improvement – To improve managerial knowledge of leaders
• Human improvement – To improve communication and cooperation, attitude to quality
• IT improvement – To support process review and plant control system

4. The company has to review, improve and rationalize the system of the company's decision-making mechanisms, as well as of the company forums and meetings. The objective of this activity is to provide that the place of decision making, as well as the person of decision maker and the relevant responsibility can be unambiguously defined, and that decisions are made at that level where the appropriate professional support is available, and that priority of the safety is provided during decision making.

5. The company has to review the methodology of safety culture surveys, including defining of actions potentially needed. The company has to perform these surveys regularly (e.g. in every two years). The scope of the surveys should cover both employees and managers of the company and should also include those of the strategic partners of the company.

6. The regular managerial assessment of the values of safety indicator system should be introduced. The company has to review the documents "Safety Policy" and "Quality Policy".

7. The company has to organize regular managerial forums at all levels of the organizational hierarchy. The human policy organisation has to elaborate a uniform consideration system for methodology of forums. The methodology has to contain constant "elements", for example safety, quality questions, as well as topical issues. The main items of these discussion topics should come from the top management, considering the most important questions and tasks of the company, but always emphasizing priority of the safety.

Regarding the safety culture indicators the current safety indicator system was developed on the basis of IAEA TECDOC-1141. Hierarchical structure of SPI system contains 4 level (72 specific indicators, 20 strategic indicators, 8 overall indicators, 3 attributes). On the top of the structure there are three main safety attributes characterizing the operational safety performance of the plant. Indicators belonging to the attribute called 'Attitude towards safety' contains indicators which in some extent can characterize safety culture. For every

	indicator goal values and thresholds for unacceptability were determined that helps to assess trends. Results of SPI assessment are presented in the Quarterly Report of SPI in the meeting of Safety and QA Management Committee. A regular managerial assessment of the values of safety indicator system has introduced. The development of a web based computer program to support the assessment work is in progress. However this indicator system is suitable for assessing the safety culture of the plant and its employees and managers as a whole. It does not assess the safety culture of individual persons.

Seq. No	Country	Article	Ref. in National Report
115	Korea, Republic of	Article 19	4.3.8

Question/ Comment	(4.3.8 Feedback, own operational experience) In section 4.3.8, it is stated that 'Event reported to the Authority are investigated at plant level, other events are investigated at professional level.' It is understood that the operating organization only investigates the serious event and that the others are not investigated by the operator. Would you explain what it actually means?
Answer	The plant experiences about 50 events that are – according to the national reporting criteria – reportable to the regulatory body. These events are investigated by an internal independent group of engineers within the Nuclear Safety Division of the plant. This type of investigation is called “investigation on plant level”. There are about 60-100 events/year, which do not meet the national reporting criteria, however – according to an internal criteria – they are also analysed. These events are not investigated by the above mentioned investigation group. They are investigated by those professional staff of the plant who were involved in the event, i.e. this is a kind of self assessment. Additionally to the above mentioned events a large number of low level events (deviations) are collected and treated by the plant.

Seq. No	Country	Article	Ref. in National Report
116	Slovakia	Article 19	4.3

Question/ Comment	Do you implement (or plan to implement) risk-informed regulation (RIR)? Do you allow a performance of the scheduled maintenance and repair during normal plant operation at full power or it is limited only during shut down of the plant?
Answer	Yes, HAEA NSD is in the process of implementing a risk-informed approach to regulation. For this purpose an Implementation Plan was prepared and approved and a comprehensive long term project was launched in 2003. The Risk-informed Implementation Project (RIP) schedules all the tasks, which have been identified important to improve the legislative, modelling and training areas as prerequisites for the successful implementation. The Paks NPP also performs scheduled maintenance during the full power operation. There are systems, which can be maintained only during full power operation state (like some special ventilation systems, which are in use only during refuelling states) and there are systems, which can be maintained beyond the refuelling maintenance period. These activities are planned for full power operation states. Some failures or breakdowns are planned to be eliminated also during full power operation.

Seq. No	Country	Article	Ref. in National Report
117	Slovenia	Article 19	section 4.3.1, p 72

Question/ Comment	Subsection 4.3.1 describes modifications of the final safety analysis report. What are the criteria considered in evaluation of licensee's application for FSAR

	modification? What is the documentation and safety assessment that needs to be provided with the application for a modification of systems, structures and components of for some organisational changes?
Answer	The FSAR that had been completed by mid 2000 have suffered from a number of deficiencies both in its contents and for its unbalanced compilation. These deficiencies have made it unsuitable for the purposes an FSAR needs to serve. The primary reason for the shortcomings was the lack of an exact and unanimous guidance for the authors of the FSAR. Therefore, besides the requirement of a review of the FSAR, the regulatory body has also ruled to prepare a QA plan for the review. The QA plan describes the generic and special requirements set for various chapters of the FSAR. The requirements include: •prescriptions on the contents •the necessary source documents •the prescriptions posed by the nuclear safety codes on the FSAR and on its selected chapters •presentation of the suitability of systems and system components •formal and editorial prescriptions •staffing needs of the review, control and scheduling. During the assessment of the reviewed FSAR, the HAEA NSD evaluates the fulfilment of the requirements stipulated in the QA plan. Content requirement of the application for licence In the application for licence, it shall be certified that safety of the unit does not decrease even during the construction phase, under the modified system or systems and system elements it is possible the unit to be safely taken in service and operated. In order to provide this, the application for licence shall contain a lot of specific data which are summarized in the Nuclear Safety Regulations. Licensing of an organization-modification The intention of a change in the organization and the management causing a result deviating from the version specified in the Final Safety Review Report shall be reported by the Licensee to the Authority at least three months before its scheduled introduction, in written form. The safety consequence shall be defined on the basis of the preliminary safety assessment and in case of a significant safety consequence, the change shall be licensed by the Authority. The licensing is executed in one stage, by means of issuance of the modification licence. Content requirements of the application for licence In the application for licence, it shall be certified that safe operation of the nuclear power plant's units is not disadvantageously affected and not decreased during execution of the organization-modification and during operation with the modified organization. In order to provide this, the application for licence shall contain a lot of specific data which are summarized in the Nuclear Safety Regulations.

Seq. No	Country	Article	Ref. in National Report
118	Slovenia	Article 19	section 4.3.1, p 73
Question/ Comment	It seems that in Hungary PSR is not required for license extension (for a predefined period of time) but for license retention. Since it is planned that lifetime (or license) of Paks NPP will be extended for another 20 years, does the regulatory body intend to consider PSR		

findings and implementation of actions to resolve the recommendations as prerequisites for extension of license?
What are other legal and technical requirements for license extension?

Answer

The legal framework of regulating nuclear safety relevant for lifetime extension is as follows: Atomic Act (Act CXVI of 1996) and the related 108/1997 Korm. Governmental Decree. They address the issue of lifetime extension as described below:

- According to the atomic act a licence (among them the operating licence) may be granted for a defined or undefined period of time, as well as subject to certain stipulations. The licence granted for a defined period may be extended when so requested.
- The Governmental Decree issued for the execution of the act clarifies that the issuance of operating license could mean the extension of the designed lifetime.
- According to the decree, so as to extend the design lifetime of the NPP units, no later than four years before the expiration of design lifetime, the Licensee shall, submit a program to the regulator, which schedules the establishment of the conditions of the operability beyond the designed lifetime. The regulator inspects the program and its implementation.
- Licensing of operation beyond the design lifetime takes place through the new operating license issued before the end of design lifetime upon the application of the Licensee. Within the procedure assessing the application the regulator considers the results of the program and its inspection findings.

Detailed regulations

Within the Hungarian nuclear regulation system the detailed prescriptions are involved into the Nuclear Safety Regulations. The regulations were issued as the appendices of the mentioned governmental decree. There are six volumes of these regulations from which the first four is related to the NPP (the other two address the research reactors and spent fuel storage facility). This four volume divide the nuclear requirements as follows:
Volume 1: Regulatory procedures,
Volume 2: Quality assurance,
Volume 3: Design
Volume 4: Operation
Originating from this fact it concludes that the regulation of different issues (for example lifetime extension) is addressed by more than one volume.

The regulation divides the lifetime extension procedure into two stages:
a) program for lifetime extension,
b) new operating license.

a) Program for lifetime extension

According to the regulation the safe operation shall be continuously maintained during the preparatory phase and during the operation beyond the designed lifetime (OBDL) in accordance with the laws and regulatory prescriptions of legal force. The problems arising from the actual operation shall be handled within the valid operating license. During the OBDL the necessary safety margins, considered by the safety analysis, shall never be consumed, not even with reference to the approaching of the end of licensed lifetime. The activity aiming at maintaining the technical conditions of the safety SSCs shall be launched and continuously performed already within the designed lifetime; additionally the efficiency of this activity shall be systematically supervised and evaluated. The determination of safety improving measures, deriving from the modern international requirements, shall be carried out within the frame of PSR and not for the lifetime extension issue.

Requirements for the program aiming at establishing the conditions for lifetime extension:
• For establishing the conditions of lifetime extension and for the justification of operability the Licensee shall prepare a program. The program and a description of its time-proportional implementation shall be submitted to the regulator no later than four years before expiration of the design life. The program can be submitted for one or more units of the same plant. In the substantiating documentation at least 20 years of operating experience shall be considered. The regulator inspects the program and its implementation (and checks for any discrepancy that could prevent licensing of lifetime extension).
• All modification and fixing activity shall be performed within the frame of the valid operating license and not in the program.
• The program shall be based on the requirements for the application of the new operating license. Here the fulfilment or status of fulfilment or the activity (with schedule) planned for the fulfilment of that requirements should be demonstrated.
• The program shall contain the planned duration of OBDL.

b) Operating license (OL)

Licensing of lifetime extension is performed in the new OL, upon the application of the Licensee to be submitted 1 year before the expiration of the lifetime. Validity: until defined time period if all conditions are fulfilled. In the OL application it should be demonstrated that:
• appropriate scoping of SSCs necessary to safe OBDL is performed;
• relevant ageing mechanisms are addressed;
• the condition of relevant SSCs are surveyed, efficiency of the former ageing programs are evaluated, new ageing management aspects and requirements are elaborated;
• scope of time limited ageing analysis (TLAA) involved in lifetime extension is determined, former TLAAs are re-evaluated and their validity is checked;
• the FSAR is actualized;
• necessary modification of operating conditions and limits are surveyed and substantiated;
• relevant documents (operating limits and conditions, maintenance policy, symptom-based emergency operating procedures, other emergency procedures, emergency response plan) are surveyed and their modifications necessary to lifetime extension are justified.
• Upon the above activities it is ensured that during extended lifetime the safety function are fulfilled at the desired reliabilty, the safety analysis covers the possible operating modes and the operating limits and conditions are in harmony with lifetime extension requirements.

The followings shall be attached to the OL application: actualised FSAR, modified version of the above documents, the necessary special authority contributions. Background documentation to the substantiating documents shall be submitted upon further regulatory request.

Re-licensing of operating and other licenses expired at the end of lifetime

Conditions for the issuance of the operating license: the temporary storage or final disposal of radioactive wastes and spent fuel shall be ensured in harmony with the international expectations and experience. The valid operating license is precondition; maximal length is

the operating license of the unit. In the application the followings shall be demonstrated:
• The operation is in accordance with the approved safety analysis.
• The inspection, manual and emergency documents and procedures are appropriate for safe operation.
• Necessary initial data for condition monitoring of the SSCs are available.
• Safe operation is ensured fulfilling the operating limits and conditions.
• Technical and administrative conditions are ensured for long term safe operation, the financial resources performing long term maintenance and development of safety are available, the possible reasons for cancellation of the license are eliminated.
• The documents and contributions needed to OL are also parts of this application.

Guidelines relevant to lifetime extension

Besides the legally binding requirements the regulator has the possibility to issue not legally binding requirements. However these regulatory guidelines has important role in the system of regulations, because if the Licensee would like to deviate from the given guideline than it shall be justified that the applied method is more or at least equally conservative than the one of the guideline. This method shall be well substantiated.

The system of guidelines follows the structure of the Nuclear Safety Regulations; that is all of them are attached to one of the volume of the NSRs. So for example concerning ageing there are 4 guidelines in which four different aspects of requirements are address. In the guidelines the requirements of the NSRs are explained in details or the method of meeting the given requirement is formulated.

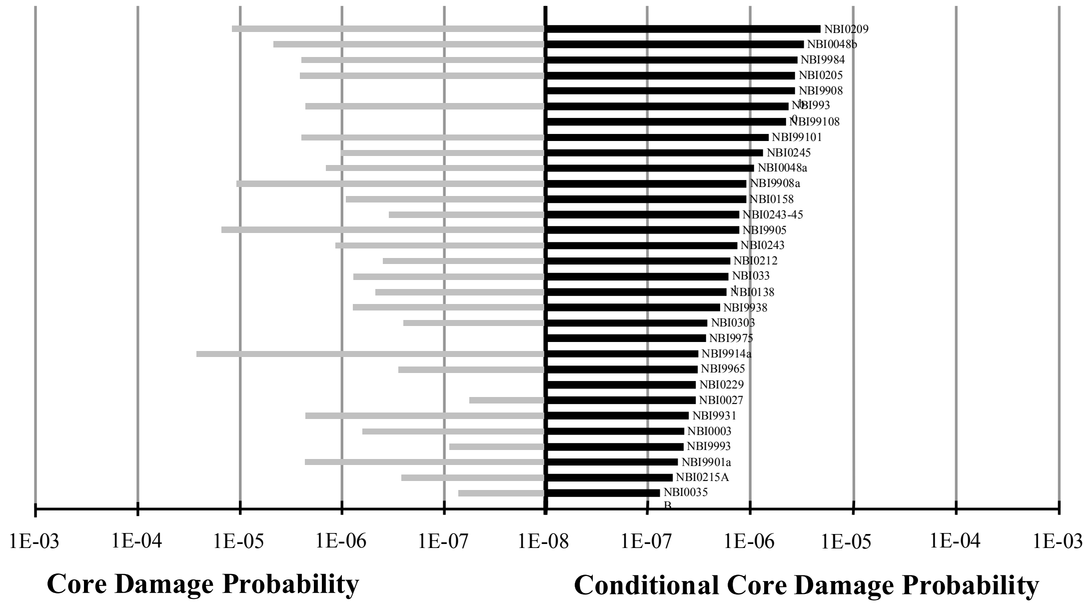
Concerning lifetime extension the following guidelines were already issued:

Maintenance
1.19 Inspection of the efficiency of the maintenance program of the nuclear power plant
4.9 Nuclear power plant maintenance program and maintenance efficiency monitoring
Ageing
1.26 Regulatory Inspection of the Ageing Management Program
2.15 Quality Assurance in the Ageing Management of Nuclear Power Plant Equipment
3.13 Consideration of Ageing during Nuclear Power Plant Design
4.12 Management of Ageing During Operation of Nuclear Power Plants
Equipment qualification
1.27 Regulatory control over equipment qualification and preservation of the qualified status
3.15 Equipment qualification requirements during the design of nuclear power plants
4.13 Equipment qualification requirement for operating nuclear power plants

Additionally the two most relevant guidelines are under issuance. These guidelines directly address the lifetime extension. The titles and numbers will be:
1.28 Requirements for the scope of the lifetime extension licence application
4.16 Conditions of operation licence renewal of nuclear installations

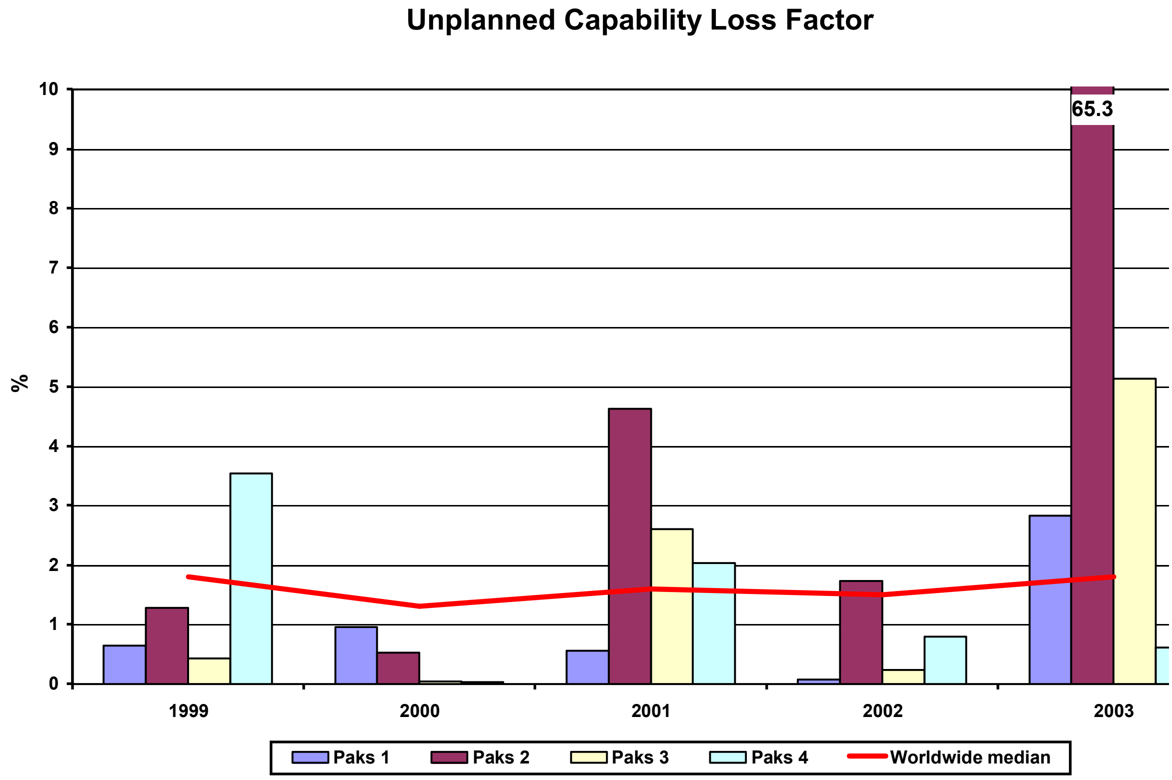
The role of the PSR:
Every ten years the Licensee shall submit a Periodic Safety Review report, which is required for license retention. The PSR gives an opportunity for a complex overview of SSCs' technical conditions, their ageing management programs, environmental conditions,

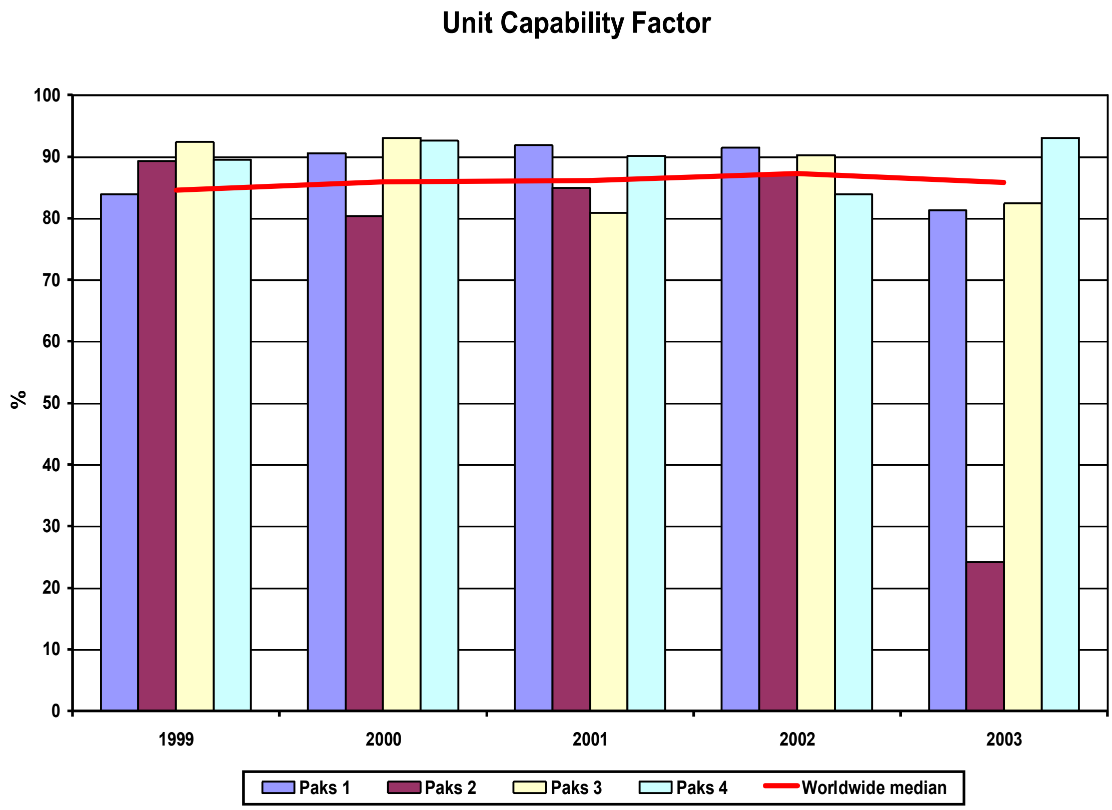
	changes in the state of the art of science and technology, utilisation of operational experience, etc.		
Seq. No 119	Country Slovenia	Article Article 19	Ref. in National Report section 4.3.8, p 78
Question/ Comment	Ta third paragraph states that since the year 2000, several events have also been analysed by probabilistic methods. Please, present some examples and results in delta CDF values. Have Paks NPP or HAEA performed also root cause analyses? Have there been observed any recurring events or events that were common to other units of the plant (generic)?		
Answer	Regular analysis of events by the Precursor Event Analysis Program started in early 1999. Since that time all events submitted to the HAEA have been subject to analysis (the current version of the event analysis tool supports the analysis of events that occur during power operation; extension to low power and shutdown states is a potential area for future improvement of the program). Events have been evaluated on a quarterly basis, and a summary of the analysis results has been put together at the end of a calendar year based on the quarterly PSA based event analysis reports. The PSA models themselves have been updated annually to reflect plant modifications and changes in component reliability data. In the period of 1999-2003 altogether 310 licensee event reports were submitted to the authority. Due to the currently limited scope of the analysis not all of these events could be analyzed. 90 of those events had no impact on the core damage risk, thus they were excluded from the scope of the event analysis. Other 71 events occurred during low power and shutdown plant operational states that are out of the current analysis scope. The remaining 149 events were analyzed. Sometimes more analysis cases had to be distinguished for one event due to the fact that, in addition to the reported event some other (e.g. latent) failures were identified on the basis of the event investigation report that had a duration different from the one reported. In such cases the additional failure was also evaluated. All together 161 different cases were analyzed. Results of the event analyses are illustrated in the attached figure, where the events are listed in the order of their importance. In every investigation the investigators are obliged to determine the root cause of the event. Paks NPP has its own Root Cause Analyses (PRCA) procedure for that purpose. IAEA-TECDOC-1278 gives a detailed description of the Paks Root Cause Analysis Procedure (PRCAP). The analyses of the root causes of an event with high safety significance as well as of those indicating deficiencies of the management and organizational culture are conducted in teamwork, with application of all the tools and techniques of PRCAP.		



Seq. No 120	Country Slovenia	Article Article 19	Ref. in National Report section 4.3.8, p 78

Question/ Comment	In the second paragraph it is stated that the power plant shows good indicators even by international comparison, as far as safety systems are concerned. Present the indicators for Units 1 to 4 of Paks NPP to allow comparison between the units. To observe the problems with the deposits on fuel, that occurred during several years, most interesting would be the indicators of load factor, forced plant shutdown factor and corrective maintenance (on feedwater and steam generators) or some other. Try to describe the indicator trends.
Answer	(See also the attached figures.) The loss of production due to the deposits cannot be observed on the trend of Unit Capability factor, because there are other factors which have more significant fluctuation in time (those are of course not part of the forced outage rate). The trend of the Unplanned Capability Loss Factor is more in correlation with the deposit problem as it can be seen on the second figure. Units 1,2 and 3 indicate a declining trend whereas unit 4 is improving on which the deposit problem did not take place





Seq. No	Country	Article	Ref. in National Report
121	France	Article 19.2	§4.3.1 - p. 73

Question/ Comment	The report states that if tasks to be completed in 2005 are not implemented it is possible that this might lead to the limitation or withdrawal of the operating license of the units. Could Hungary specify which among these measures would lead to a license withdrawal?
Answer	Withdrawal of the operating license might have been considered by the regulatory body if any of the safety increasing measures were not initiated or suffered such a delay that the safety risks originating from that would be unacceptable. 13 out of the 15 safety increasing measures have been completed by the end of year 2004. The measures no.1 an no.14 are in a state that they shall be completed during this year. The operating license of the units of Paks NPP had been limited once by the nuclear authority when realising deficiencies in the seismic resistance of the facility. The license has so far never been withdrawn.

Seq. No	Country	Article	Ref. in National Report
122	Germany	Article 19.2	p. 75, 4.3.2

Question/ Comment	Is there a procedure for updating the „limits and conditions for safe operation“ (TechSpec) according to operation experience feedback and the state of the art?
Answer	Yes, the Guideline 4.2 Operational Limitations and Conditions (OLC) plays that role. This document describes formal requirements (place of OLC in the plant documentation system, extension, manageability of OLC etc.), general content requirements (purpose, construction) and detailed content requirements (kind of limitations by operational modes, systems and equipment - e.g. active zone). Both at the utility and the authority there are responsible units or persons dealing with the regular maintenance of the OLC. During plant or system modifications a basic task is to investigate the impact of the modification on the OLC. If the OLC shall be changed it is initiated within the licensing procedure of the given modification. Each changes of the OLC shall be approved by the HAEA NSD. The Utility has its own Procedure but this document in not in the direct sight of HAEA NSD which means that the changes or the content of the document are not governed by the HAEA NSD. Of course as in the case of any documents of the utility the HAEA NSD has an indirect influence on this procedure through the regular inspection activities (remarks, recommendations etc.).

Seq. No	Country	Article	Ref. in National Report
123	Germany	Article 19.4	

Question/ Comment	Is there a standardised structure and procedure for validation, training and application of Emergency Operating Procedures and SAMG?
Answer	There are specifications for modifications, validation, application and training of the EOPs. The structure of the EOPs corresponds to the structure of the symptom oriented procedures by Westinghouse. The HAEA NSD has approved these procedures and specifications. Elaboration of the Severe Accident Management Guidance begins in this year and will be completed by the end of 2007. Its structure will also correspond to the Westinghouse documentation system. At the beginning of the project a QA plan shall be elaborated, which shall lay down the V&V requirements and the requirements of documents referring to the application and training requirements.

Seq. No	Country	Article	Ref. in National Report
124	Germany	Article 19.4	p. 75, 4.3.4

Question/ Comment	Which further safety improvement measures are prepared or planned in connection with severe accidents? Is there an Accident Management Program for prevention and/or mitigation of severe accidents?
Answer	At the end of 2004 a strategy of accident management procedures was submitted to HAEA

NSD containing a set of tasks to be solved in the near or later future. Up to now the first version of the document has been discussed by the Paks NPP, HAEA NSD and AEKI (academic research institute, author of the strategy). After some corrections it has been agreed and a resolution has been issued that a detailed schedule of the realisation shall be ready by the end of 2005.

The most important tasks are

- At a short deadline a seismic study has to be done and its results shall be taken into account in the strategy.

- A symptom-based Severe Accident Guideline (SAG) has to be elaborated. This SAG will be available in printed form and on the process computers of the NPP and it will have the appropriate interfaces to the existing Guideline for DBAs.

- The existing Guideline for DBAs shall be completed for the shut-down states of the reactor and the events at the reactor pool.

- A thorough revision of the minimum set of measurements in accident situation shall be established.

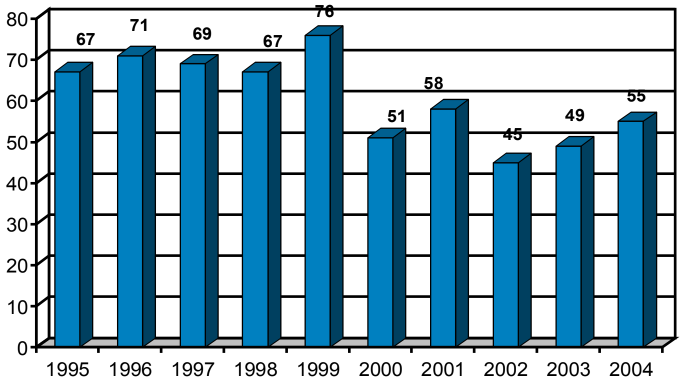
- A list of technical measures: sizing of hydrogen recombiners, studying of the effectiveness of the Sprinkler system, flooding of the reactor shaft in case of the failure of the reactor vessel • ¨ strengthening of the maintenance door of the shaft, examination and if necessary modification of the electric power supplies, etc.

- Organisational changes

The existing Guideline for DBAs contains preventive instructions to avoid severe accidents that start with core melting. The mitigation of the consequences of severe accidents is expected to achieve by the instructions of the SAG.

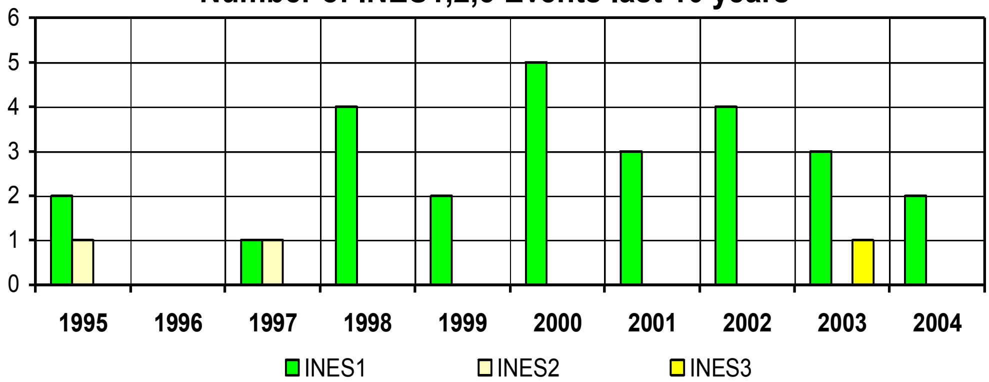
Seq. No 125	Country Finland	Article Article 19.7	Ref. in National Report chapters 3.4.1; 3.2
Question/ Comment	Benefiting from the operating experience feedback is dealt with in several places in the report, such as in 3.1.4, 3.2 and 4.3.7-4.3.8, as an important contributor for safety. We would like to ask to show safety indicators such as 1. Number of reportable events from PAKS NPP to the regulatory body, 2. Number of INES classified events and 3. Number of IRS reports written during the last ten year period. Has analysis of events and feedback from the ten year period shown any interesting technical or human/organizational factors (e.g. in addition to Annex 7) from which other organizations, e.g. VVER users, can learn?		
Answer	On the attached two figures (see Support Document "Safety indicators") the number of reportable events to the regulatory body and the number of INES classified (1,2,3) events are shown for the last 10 years. The total number of the IRS events reported to the IAEA is 13.The selection of the events for reporting indicates the fact that those may be of interest for the international nuclear community. For example the event with the deposits on the fuel assemblies and the event with the damage of fuel assemblies on unit 2 are among those reported to the IAEA.		

Number of Safety related Events reported immediately to the regulatory body last 10 years



Year	Events
1995	67
1996	71
1997	69
1998	67
1999	76
2000	51
2001	58
2002	45
2003	49
2004	55

Number of INES1,2,3 Events last 10 years



Year	INES1	INES2	INES3
1995	2	1	0
1996	0	0	0
1997	1	1	0
1998	4	0	0
1999	2	0	0
2000	5	0	0
2001	3	0	0
2002	4	0	0
2003	3	0	1
2004	2	0	0

Seq. No 126	Country France	Article Article 19.7	Ref. in National Report §4.3.8 - p. 78
Question/ Comment	Could Hungary indicate if significant additional lessons were learnt from events analysed by probabilistic methods?		
Answer	Several lessons have been learnt from events analysed by probabilistic methods: ƒ{ In 1997 the new Nuclear Safety Codes have introduced new and very strict reporting requirements on events at the Hungarian nuclear installations. Due to this the number of events reported to the HAEA increased significantly. After the evaluation of the events		

using the PSA based precursor methodology it was revealed that the significance of the majority of the reported events was much less then what the regulatory attention would require. Based on this conclusion the reporting requirements were revised.

ƒ{ An assessment was performed to compare the INES rating of the events and their safety significance determined by probabilistic methods. The comparison could not clearly define coherence between the two ways of judging the significance of the events. It was concluded that much subjective elements were taken into consideration when the INES rating was performed. As a result a more clear internal procedure was developed for INES rating of the events.

ƒ{ Using initial probabilistic event assessment the events are ranked by risk significance categories; these categories help to adequately define the proper regulatory resources to be devoted to further investigation of the event.

ƒ{ The results of the probabilistic event assessments have been taken into consideration in the annual safety performance evaluation of the Paks NPP. The severity of the events and the core damage index of the units have been followed year by year to trend any major change in the safety performance.

Seq. No	Country	Article	Ref. in National Report
127	France	Article 19.7	§4.3.8 - p. 79

Question/ Comment: Could Hungary indicate the status of implementation of recommendations issued from the analysis of the Paks event?

Answer: Most of the corrective actions that should be implemented after the April 2003 Paks event have been carried out both by the NPP and the HAEA. Smaller part of the actions (for instance organisational and functional improvement program) is in progress and the majority is expected to be finished by the end of this year. A recent follow-up mission from the IAEA has concluded that 71% of the recommendation proposed by an IAEA expert mission in June 2003 has been fully resolved, 29% of the recommendations exhibited sufficient progress, and there was no such recommendation where insufficient progress was experienced.

Seq. No	Country	Article	Ref. in National Report
128	France	Article 19.7	p. 82-83

Question/ Comment: The report mentions the improvement measures decided after the incident on 10 April 2003 together with the modification of the Final Safety Analysis Report expected to be finish in 2004. Could Hungary highlight the main modifications in this FSAR and their status?

Answer: Only those changes are introduced into the FSAR, which may have longer term influence on safety or on systems related to safety. The version completed by the end of 2004 does not yet reflect any major change performed following the 2003 April incident. The related necessary FSAR changes shall first be described in conjunction with the application for license in principle for the recovery of damaged fuel elements. Assessment and approval of the proposed FSAR modifications shall be performed by the authority parallel to the evaluation of the license application and shall be included into the new version of the FSAR dueto be submitted next year. These modifications shall reflect those changes of the plant technology and circumstances that are long standing and result from the incident and/or the recovery, such as change in the composition of the liquid waste, increase of amount and change in the composition and distribution of the stored radioactive wastes, possible effects of changes on the decommissioning.

Related to the incident in April 2003, the Paks NPP has performed a number interventions in order to improve its functioning, however these have no effect on the FSAR

Chapters and subchapters of FSAR likely to be affected are:

9. Other systems and buildings, constructions
9.1 Fuel storage and management (temporary storage of damaged fuel elements in the spent fuel pool)
9.5. Ventilation and air conditioning systems (filtered venting of the reactor hall)
11. Radioactive waste management (new waste composition, separated storage of waste related to the fuel damage and devices necessary for that)
18. Preliminary plan for decommissioning of the nuclear power plant and its units

Seq. No 129	Country Germany	Article Article 19.7	Ref. in National Report p. 78, 4.3.8
Question/ Comment	The operational experience from incidents, maintenance and in-service inspection is considered in the simulator training in order to distribute the lessons learnt to the plant operators. Are there also other methods for distributing important operational experience to the plant personal for strengthening the prevention level?		
Answer	The shift personnel is trained about both the internal and external operational experience not only in the frame of simulator trainings but also on special or refreshing classroom trainings. Additionally to this trainings operating experience information is disseminated to the technical staff of the plant using classroom training, group discussions and through the internal computer system (intranet).		

Seq. No 130	Country Germany	Article Article 19.7	Ref. in National Report p. 80, 4.3.8
Question/ Comment	Based on the experience of the Unit 2 event (April 2003), Paks NPP intents to invite international review teams every 2-3 years. Has Paks NPP implemented safety management measures as a consequence of the lessons learnt from the Unit 2 event?		
Answer	Yes it has.		

In order to identify the weaknesses in operation of the organization, first an Organisational Diagnosis was performed in the second half of 2003. This study analysed categories of culture, organisation, persons and system (of operation) and evaluated weaknesses in the past using internal and external review results. As the result of the study, a decision on the implementation of the Program of Organisation Development (ODP) was made. The goal of this safety management program is to enhance safety culture and to improve safety generally at the plant.

The ODP consist of 5 areas for improvement to solve the problems identified in the Diagnosis:
• Vision and values – To review the corporate goal system
• Optimisation of functioning of the organisation – For exact specification of responsibilities, to enhance decision making and to improve quality system and plant operation generally
• Leadership improvement – To improve managerial knowledge of leaders
• Human improvement – To improve communication and cooperation, attitude to quality
• IT improvement – To support process review and plant control system

1. Vision and Values
During the year 2004 the corporate Vision was revised. It states that the most important goals is safe, economic and long term electricity generation. The document of Values declares importance of commitment to nuclear safety, regulated operation with good

cooperation of organizations, competitive business management, committed employees and managers.
Based on the Vision and Values the basic corporate Strategy for the 2004-2014 period was elaborated. The strategic goals (21), indicators and actions were defined in five strategic areas (Safety, Business and Effectiveness, Market and Stakeholders, Products and Services, Operation Improvement).
Now preparation of the functional strategies is going on. The finalization of the plant indicator system and a set of the strategic risks is under way.

2. Optimisation of functioning of the organisation
In order to be able to specify clear responsibilities, review of all processes and procedures has been started. The software was selected and operations regulations were reviewed. A new plant Quality Management Regulation was prepared, which will be issued after review of the Nuclear Safety Regulations.
As quick tasks, review of meetings and system of managers' supervision was conducted in order to support the work of the managers. A new guideline was issued in order to improve effectiveness of meetings. Also a new procedure was issued for managers' supervision, based on the Diagnosis, internal needs and external experience.
Implementation of a comprehensive plant risk management system was initiated. As a result, risk management systems were elaborated for Labour Safety area, maintenance and economics. For operations an interaction analysis system was implemented.

3. Leadership improvement
Leadership improvement is based on corporate strategy and values and – beyond knowledge and skills - it is focused on practical use. Delivery of an intensive, short-term training course has been started that addresses those knowledge of the managers that are necessary for safe daily management. Participants of this training course are the middle level managers. The most important competencies are: safety consciousness, strategic thinking, commitment, change management.
Concept of a complex leadership improvement program (Paks Academy) was prepared objective of which is to create a system of leadership improvement that addresses the establishment and maintenance of management-specific knowledge, skills and attitudes in a complex manner as well as ensures both the transfer of knowledge and appropriate coaching. Participants of the training course are the top- and middle level managers, supervisors with direct operative control and the members of the Bank of Talents for future Managers.

4. HR improvement
In the framework of the HR improvement, an assessment of the employees' commitment was performed and based on the results area specific improvement tasks were specified.
The performance planning and appraisal system was revised to meet plant needs to integrate strategic goals and to have a more comprehensive and differentiated evaluation of performances and a more efficient appraisal of the employees.
A concept for adaptation and mentoring system for new entries was elaborated to meet company need of high-level educated and trained personnel in relationship with life-time extension program of the plant.
The development of the mental support system covered total revision of the former psychological support system. All licensed employees participated in discussion in order to define their mental condition. After process of data managers received training and support in order to evaluate the results. This kind of support will be extended and regularly

provided.

5. IT improvement

The main task of this subproject is to support the most important strategic tasks of the plant. One of the major tasks is Management / Company Information System (CIS) to implement Strategy controlled /based management.