ANSWERS TO QUESTIONS

ON

NATIONAL REPORT OF

THE SLOVAK REPUBLIC

COMPILED ACCORDING TO THE TERMS OF THE CONVENTION ON NUCLEAR SAFETY

BRATISLAVA
APRIL 2005

QUESTIONS

Austria

1. The introduction of the Report defines the strategic goal as to "sort out the rear nuclear fuel cycle concept". Please, explain strategic goal.

   Response:
   The text ´sort out the rear nuclear cycle concept should be deleted (only two bullets remaining).

2. The Report states that a set of accident analysis for Bohunice V-1 has been accomplished. Was it possible to be prove that a double ended Loss of Coolant Accident (LOCA) 2-F-500 is now Design Basis Accident (DBA) for these units, or is this still a Beyond Design Basis Accident (BDBA)?

   Response:
   In the original design of V-1 the double ended LOCA 500 mm was a beyond design basis accident without specific acceptance criteria. In accordance with the Decision No. 1/94 of the Nuclear Regulatory Body the ECCS, Accident Localisation System and supporting systems have been upgraded in the frame of the extensive safety upgrading project of V-1. The design of the upgraded systems was optimised with respect to acceptance criteria. In the current status of the V-1 plant:
   - The new Maximum Design Basis Accident after the safety upgrading is the double ended LOCA 200 mm
   - Double ended LOCA 500 mm remains to be classified as beyond design basis accident, but specific acceptance criteria have been defined by the regulatory body (together with the analytical methods to be used)
   - Acceptance criteria applicable for double ended LOCA 500 mm are the same as those for design basis accidents
   - Verification of the compliance with the acceptance criteria in case of Double ended LOCA 500 mm is based on realistic (best estimate) analytical methods (approach generally applied for beyond design basis accidents)
   - Verification of the compliance with the acceptance criteria for all design basis accidents is based on conservative methods.

3. The Report proves for BohuniceV1 unit 1 that the Level 1 PSA CDF at low power and shutdown is higher than the CDF for full power. Similar are results for V-2 unit 3 PSA study. A Level 2 PSA study was completed for full power and shut down reactor in June 2003 on the reference NPP V-1 unit 1. Have these studies been verified by an independent technical support organization and validated by the regulatory authority?

   Response:
   Level 1 PSA studies for full power, low power, and shutdown of Bohunice V1 and V2 NPPs were reviewed by IAEA missions, private foreign companies, and by UJD. The procedures applied by UJDin assessing these studies are in accordance with requirements described in Review of Probabilistic Safety Assessments by Regulatory Bodies, Safety Reports Series No. 25, IAEA, Vienna, 2002. UJD has accepted these studies.

4. What is the status of the implementation of safety improvement measures that are planned for 2004 at the NPP Bohunice V-2?

   Response:
   The measures have been implemented according to time schedule of NPP V2 Upgrade and Safety Improvement.

5. Has the level 1 full power PSA at the NPP Mochovce units 1 and 2 been verified by an independent technical support organization and approved by the regulatory authority?

Response:
The Level 1 for full power PSA study for the NPP Mochovce units 1 and 2 has been independently verified and accepted by UJD.

6. The value 1.66E-04/year for CDF for low-power and shut down reactor at the NPP Mochovce units 1 and 2 requires immediate actions. Which measures will be taken to improve safety?

Response:
The Mochovce NPP initiated a corrective measure to improve the procedures (Symptom Based Emergency Operating Procedures for a Shutdown States - SD EOPs) for shutdown reactor operation. The project is managed by BNFL on behalf of UK Department of Trade and Industry within Nuclear Safety Programme. The development of the SD EOPs is performed in cooperation with Westinghouse Electric Europe sprl. Brussels, Belgium. The project covers the development, validation and training material. The end of 2005 schedules the duration of the project.

7. A new cooling water system for pool water has been installed in the interim spent fuel storage facility. Which measures have been taken to prevent LOCAs (loss of coolant accident) in the spent fuel pool’s cooling system?

Response:
A rupture of the cooling water system pipeline can be regarded as LOCA in terms of pool cooling. Handling such an emergency is provided by the technical solution to the cooling water system. The water intake pipeline of the storage cooling pool is 400 mm below the level. A siphon is installed on this pipeline bleed the whole system in case that the level drops by 400 mm, thereby stopping a leak.
The inlet of the cooling water and its dispersion takes place at a depth of about 400 mm below the level. In case of a rupture of the inlet pipeline, the water is drained off to a depth of up to 400 mm from the nominal level.
In the course of such accident the radiation situation does not change substantially both inside the interim spent fuel storage facility and at its surroundings.

8. What kind of accident was the basis for determining the limit for the operator’s financial liability in the event of a nuclear accident? How does the operator ensure that he has the necessary funds for damage compensation ready?

Response:
At the determination of liability for nuclear damage of the licensee no concrete nuclear event was presumed. It was based on the minimum liability limit laid down in Vienna Convention from 1963. With reference to void the risk of the exchange rate differences between the Slovak Crown (SKK), Euro (EUR) and US Dollar (USD) as well as void the situation when we shall fall under the limit defined by the Vienna Convention, the limit was increased respectably and laid down in Euros. The licensee is liable to cover liability by insurance or some other form of financial cover pursuant to the § 30 Sec. 1 of Act. No. 541/2004 Coll. II.
New Atomic Act No. 541/2004, which entered into effect on 1 December 2004, has set a new maximum liability limit 75 mil. EUR. For the time being, SE is in the process of preparation of a new insurance policy with Slovak Nuclear Insurance Pool for this new liability limit. The process is close to a signature of the contract.

9. Is a double ended guillotine break of the primary pipe with major diameter under conservative conditions considered as DBA for all operating reactors in the present Slovak Nuclear Legislation?

Response:
The Legal basis contains requirements concerning the scope and extend of the safety documentation required by the regulatory body for the review and assessment during licensing of the nuclear installations (including requirements for the list of the Postulated Initiating Events (PIEs) included in the design). The specific list of PIEs involved in DBA is not stipulated and it is the responsibility of the licensee to determine this list of PIEs in an early stage of the design development.
UJD is responsible for the review, assessment and for acceptance of the safety level of nuclear installation during the licensing process.

For all of the WWER 440/V213 type reactors in the Slovakia the double-ended guillotine break of the primary pipe with the largest diameter is a part of DBA, analysed by a conservative approach.
For the first generation of WWER 440 units with V-230 reactor type this event was not considered in the original Russian design (in the list of DBA PIEs). However, this postulated initiating event was taken into account during the gradual safety upgrading of these units in the Slovakia, using the best estimate approach in the accident analysis.

10. Has the Regulatory Body established a Quality Management system? On which approach or model is this system based? Are audits or external reviews regularly performed? Has the Authority organized a team or group independent of regulatory activities to perform internal audits and self-assessments? If so, how many staff persons are in the independent team or group?

Response:
UJD has worked-out the Quality Management System based on the STN EN ISO 9001: 2000, IAEA 50-C/SC-Q, and IAEA TECDOC–1090. Beside regular internal audits done by UJD's internal Audit Department, additional reviews of this system have been accomplished by external agencies, including two reviews by IAEA, using its International Regulatory Review Teams. The additional reviews have been done by the following agencies – the Supreme Audit Office of the Slovak Republic, the Government Office, and the Ministry of Finance of the Slovak Republic.
UJD starts to apply (first departmental evaluation was done 2003) also the EU public administration's Common Assessment Framework (CAF). CAF integrates the principles of the models from the European Foundation for Quality Management and from Speyer, the German University of Administrative Sciences. Appointed team of 8 staff persons works to perform full internal audit and self-assessment for the whole UJD now. An implementation of CAF in Austria is managed by The Federal Chancellery, Dept. for Administrative Reform, Wollzeile 1-3, A - 1010 Wien, Tel.:+ 43-1-50190-7148.

11. Which measures have been taken or are planned to ensure that institutions working on behalf of the regulatory body do not work for the operator's institution and are sufficiently independent?

Response:
Any institution in safety assessment evaluation business could take part in it only, and only when it has implemented acceptable internal Quality Assurance Management. That system assures that institutions working on behalf of the regulatory body do not work for the operator's institution and are sufficiently independent.

12. The description of the operator's responsibilities discusses only issues of regulation and inspection. Is the prime responsibility of the operator clearly defined and the license holder committed to the Convention's goal regarding primary safety responsibilities?

Response:
The prime responsibility for nuclear safety of the licensee is clearly defined in the Atomic Act (No 541/2004), §23, art. 1. Detailed description of the licensee's responsibilities during the operation of the nuclear installation are given in the Atomic Act, sections IV, VI and others.

13. The Report states that the period between 2001 and 2003 featured high financial intensity of debt servicing and unbalanced financial standing in the area of resources and demands, i.e. incapability to repay the debt service through SE a. s. own resources. How will this issue be resolved to reduce the risk for safety at the NPP due to insufficient financial resources?

Response:
The period between 2001 and 2003 was characterised by a high intensity of the debt service. As from 2003 SE, a. s. , was focused on systematic work with the banks with a view of gaining their confidence. SE, a. s. , efforts were aimed to revitalise the economic situation and fix its capability to repay debts. These efforts resulted in awarding the BB+ (stable) rating by Fitch Ratings Ltd. as compared to BB+ (negative) over 1999-2003.
Under the company financial revitalisation process, a state-non-guaranteed debt restructuring was undertaken in 2004, on which SE, a. s. , acquired resources totalling € 350 m. The following credits were early repaid that year: VÚB SKK542 m., Interbanka CZK 600 m, VÚB SKK 700 m, Bank of Tokyo € 75 m, EBRD SKK 5 bn. These loans were replaced with less-expensive resources, namely a €330 m open-end credit from WestLB AG (we currently report its zero status), a € 30 m credit from WestLB AG.
State-guaranteed debt restructuring Phase 2 is under way in 2005. Resources totalling € 500 m have been provided to this end.
The above restructuring process has largely concerned EMO debt service, as EMO accounts for 86% of the total status of SE, a. s. , bank loans.
The aim of the debt restructuring is to optimise cash flows over the following periods and improve the credit conditions, i.e. replace the old debts bearing a high interest rate with new more favourable credits. By spreading the due dates SE, a. s., financial risk has reduced.

14. The Report states that the preferred decommissioning option is a very long delay between shutdown and decommissioning of the facility. In which way will be guaranteed that the required technical and scientific workforce as well as the necessary funds for financing be available at the site of NPP Bohunice V-1, if decommissioning would start in 2038?

Response:
It is expected that a new entity will be established in the near future (2005-2006) to be in charge of nuclear installation decommissioning. Since this is basically the current SE-VYZ, there is a prerequisite for maintaining the high professional standards of all the activities relating to the shutdown and decommissioning of nuclear installations at the Jaslovské Bohunice site. The need for funds, as to date, will be covered through State Fund for Nuclear Installation Decommissioning.

15. A private company is preparing a plan for the privatisation of Slovenské elektrárne, a. s. How will the necessary funding for the decommissioning of the NPPs Bohunice A-1 and V-1 under these new circumstances be ensured?

Response:
The changes in the organisational structure of the privately owned company operating nuclear sources should substantially not affect the required funds for decommissioning the NPPs A-1 and V-1, which should continue to be financed from the State Nuclear Decommissioning Fund. The replenishment of the Fund is principally affected, and this is going to be in the future as well, by the amount of contributions by the operators. The sufficiency of these funds (the amount of contributions) will be secured through amendments to the current legislation (State Fund Act, etc.)

16. What are the qualification requirements for contractor’s personnel? How are these requirements supervised? Who is responsible to review the contractor’s QA?

Response:
According to the Act No. 541/2004 (Atomic Act) the licensee is responsible for nuclear safety. The licensee is also responsible for all the works at nuclear facility done by any contractor.
Any contractor which do any work at nuclear facility have to have its own Quality Assurance system. This QA system shall be in compliance with EN ISO 9000. At the selection process, the supplier (contractor) submits to the licensee an offer for the required work and one of the documents as a part of the offer is a certificate on QA system issued by an Auditor.
This mean that the contractor shall have an appropriate organizational structure, qualified managing personnel, specialised personnel structure particularly for the work to be done, documentation and record keeping system, appropriate technical resources (machines, equipments, shops), etc.
The qualification requirements are usually contained in the contract (e. g.):
- requirements for a special technical education and its level (e.g. skilled workman, secondary technical school, university degree)
- training and certificate on a special qualification required by generally binding legal regulations and technical norms (e. g. welders, crane operators, non-destructive inspection personnel, etc.)
- practical experience of the personnel in the job to be done
- personnel health certificates for jobs in the radiation envinronment, etc.

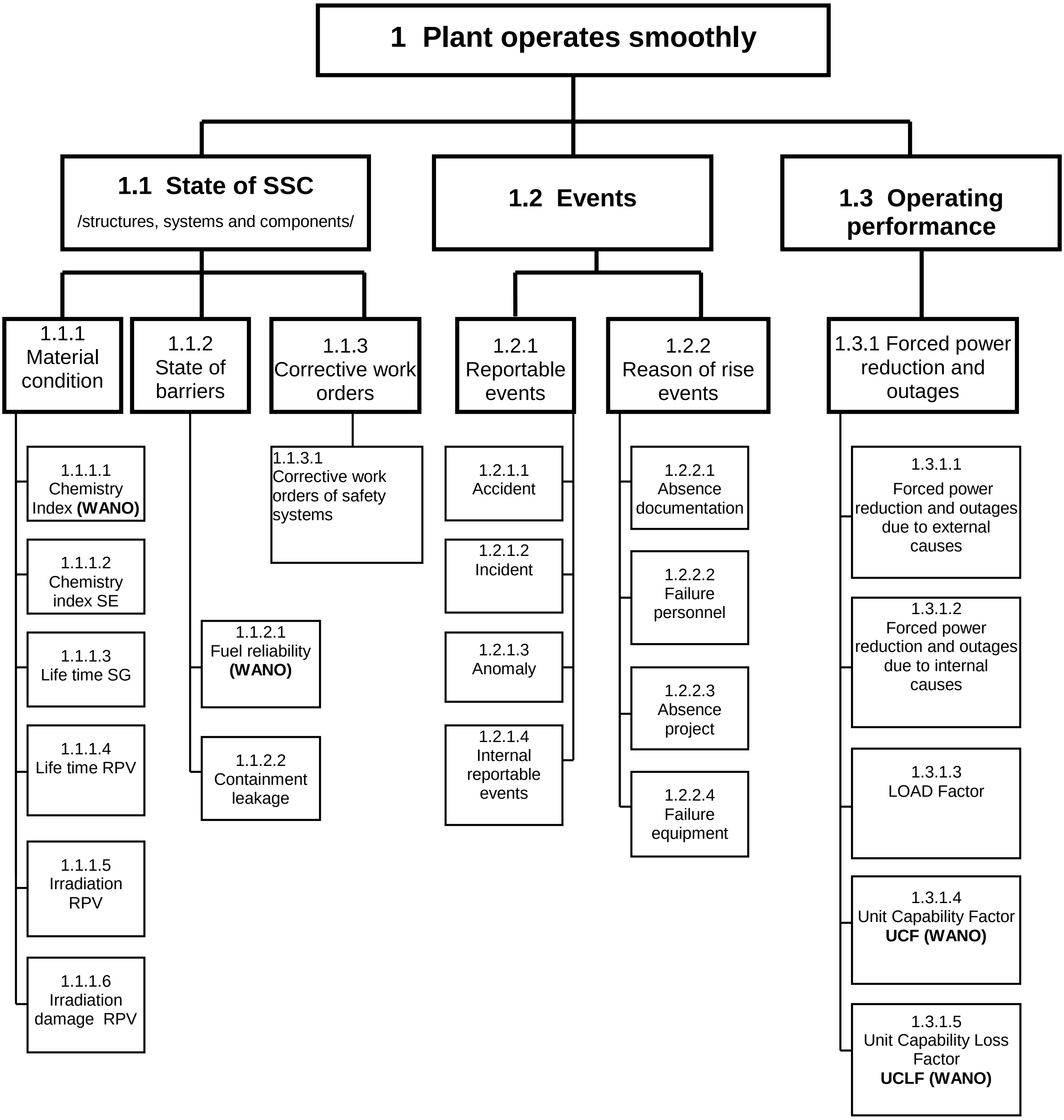
The responsibility for the control and supervision of qualification requirements is on the head of the organizational unit of the operator responsible for the relevant area. He (she) is responsible for the checking of associated personnel educational certificates, job supervision, post job testing supervision, documentation supervision, etc.
UJD controls the operator´s activity in the field of supervision of contractors and their personnel and the compliance of these activities with the operator´s QA system. The QA system of the operator have to be in compliance with § 4, article 2, a) 2 of the Atomic act and approved by the decision of UJD. UJD also performs inspections during outage activities and there is a possibility to check the activities done also by the contractors.

17. The Report describes that the NPPs are monitoring a set of performance indicators in accordance with IAEA documents. However, in chapter 6.1.2 only Unit Capability Factor and Load Factor are shown. Could you present similar figures for safety performance indicators showing Safety Systems Performance and Tightness of physical barriers?

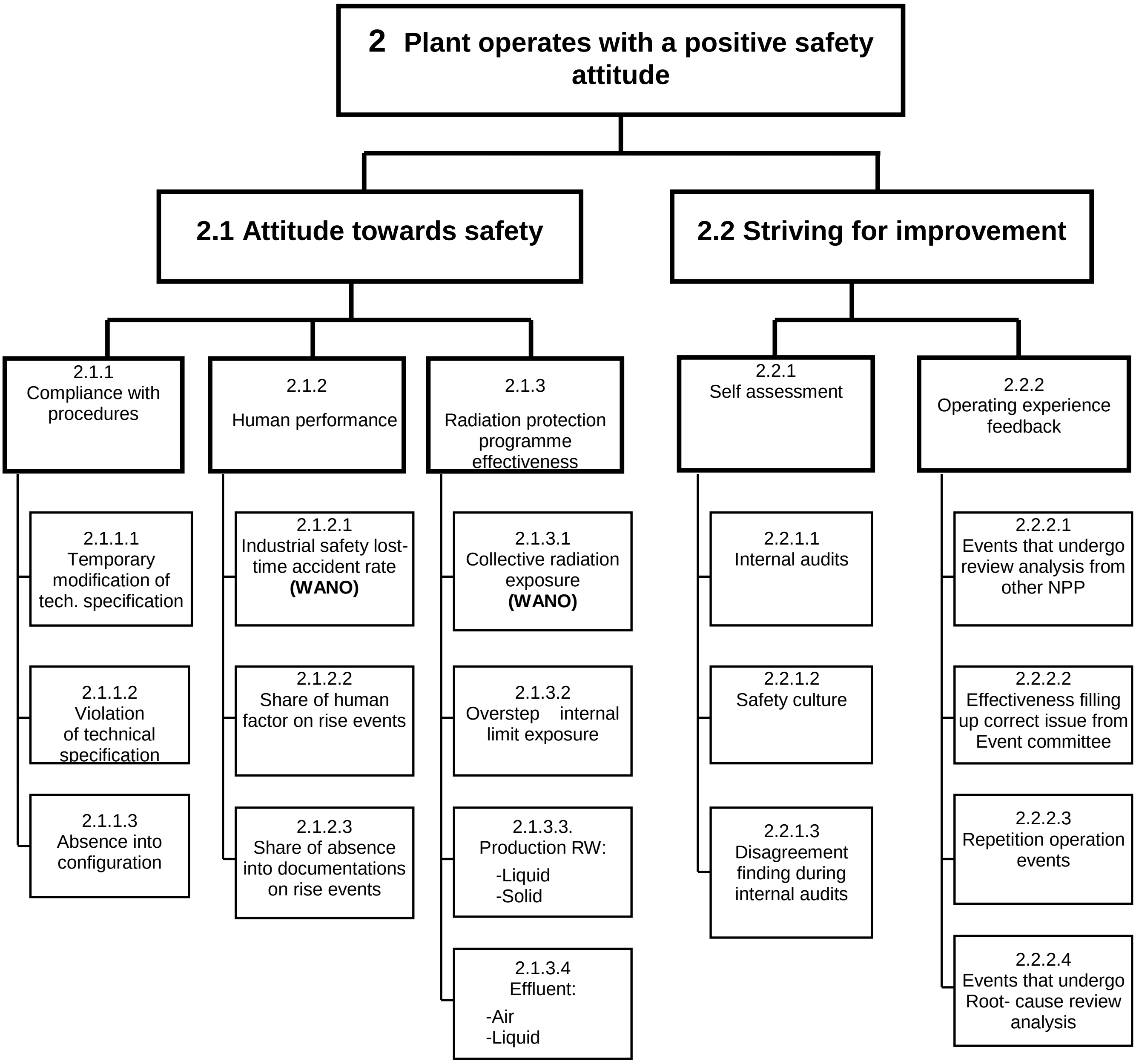
Response:
The operational safety performance indicators (OSPI) system is used at SE, a. s., to assess the nuclear installation operational safety in accordance with the basic structure defined in Annexes B, C to SE/MNA-051.01 - Assessment of Nuclear Installation Operation Safety.
The system has been developed in line with IAEA - TECDOC -1141 “Operational safety performance indicators for nuclear power plants‟ and TECDOC - 1125 ″Self -assessment of operational safety for nuclear power plants‟.
The OSPI calculation methodology is described in SE/MNA-051.01-01- Operational safety performance indicators.
The software PPRC - Self -assessment of operational safety for nuclear power plants - is used to collect and record data and work with OSPIs at IIS-SE.
The entire process of work with PPRC is described in the user’s manual ‟Application of PPRC, version 1.83, for self-assessment of operational safety for nuclear power plants ‟
The state of operational safety of SE-EBO, EMO units, and also VYZ units - has setup other indicators - is periodically evaluated, at least on a quarterly basis by analysis of OSPIs and meeting the goals.
SE, a. s., HQ prepares the annual ″Complex Report on Nuclear Installation Operational Safety‟

SE, a. s., HQ co-operates with special departments at the plants in PPRC upgrade.
The entry of WANO "Operational Indicators" input data is ensured in accordance with the requirements of the software.
The software has been installed centrally at SE, HQ.
The OSPI structure for operational units is shown in the following annexes.

SE structure of SPII



SE structure of SPI



Identification Card for Specific Indicator in the Safety Performance Indicator system - year 2004

1. Title:	**Unit Capability Factor - UCF** /WANO /	**2. Units:**	SE-EBO 1.V-1

3. Purpose:

The purpose of this indicator is to monitor progress in attaining high unit and industry energy production reliability

4. Definition:

Indicator is defined as ratio of the available energy generation over a given time period to the reference energy generation over the same pe time period, expressed as a percentage.

Calculation:

$$UCF = \frac{(REG - PEL - UEL)}{REG} \times 100 \quad (\%)$$

REG - reference energy generation for the period
PEL- total planed energy losses for the period
UEL- total unplanned energy losses for the period

5. Data and requirements for data collection:

6a. Acceptable boundary:	**82%**	**6b. Standard - Plan:**	**86%**	**6c. Strategic goal:**	**89%**

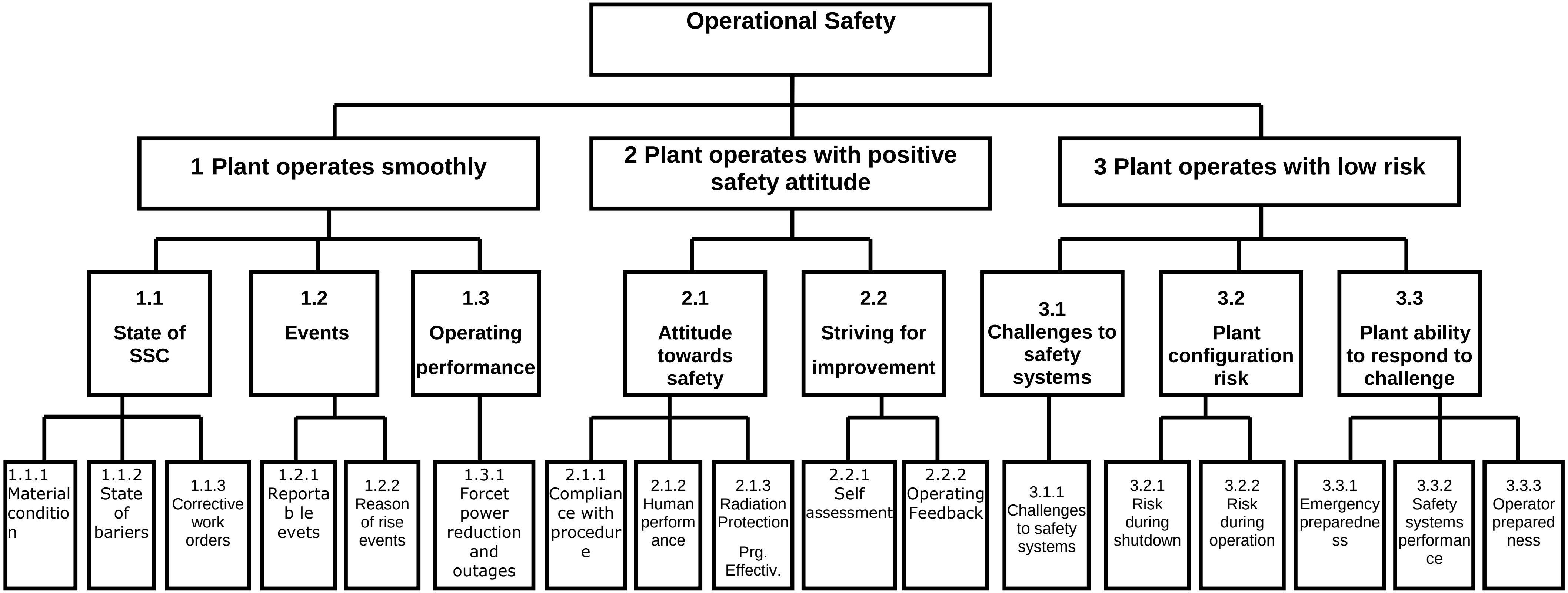
7. Format of values:	**X Integral number** ☐ **decimals** ☐ **exponent**	**8. Value measure:**	Min: 0 % Max: 100 %

9. Min period:	☐ **Weak** ☐ **Month** **X Quarter** ☐ **Year**	**10. Begin:**	2000	**11. Guarantee - centre:**	27060

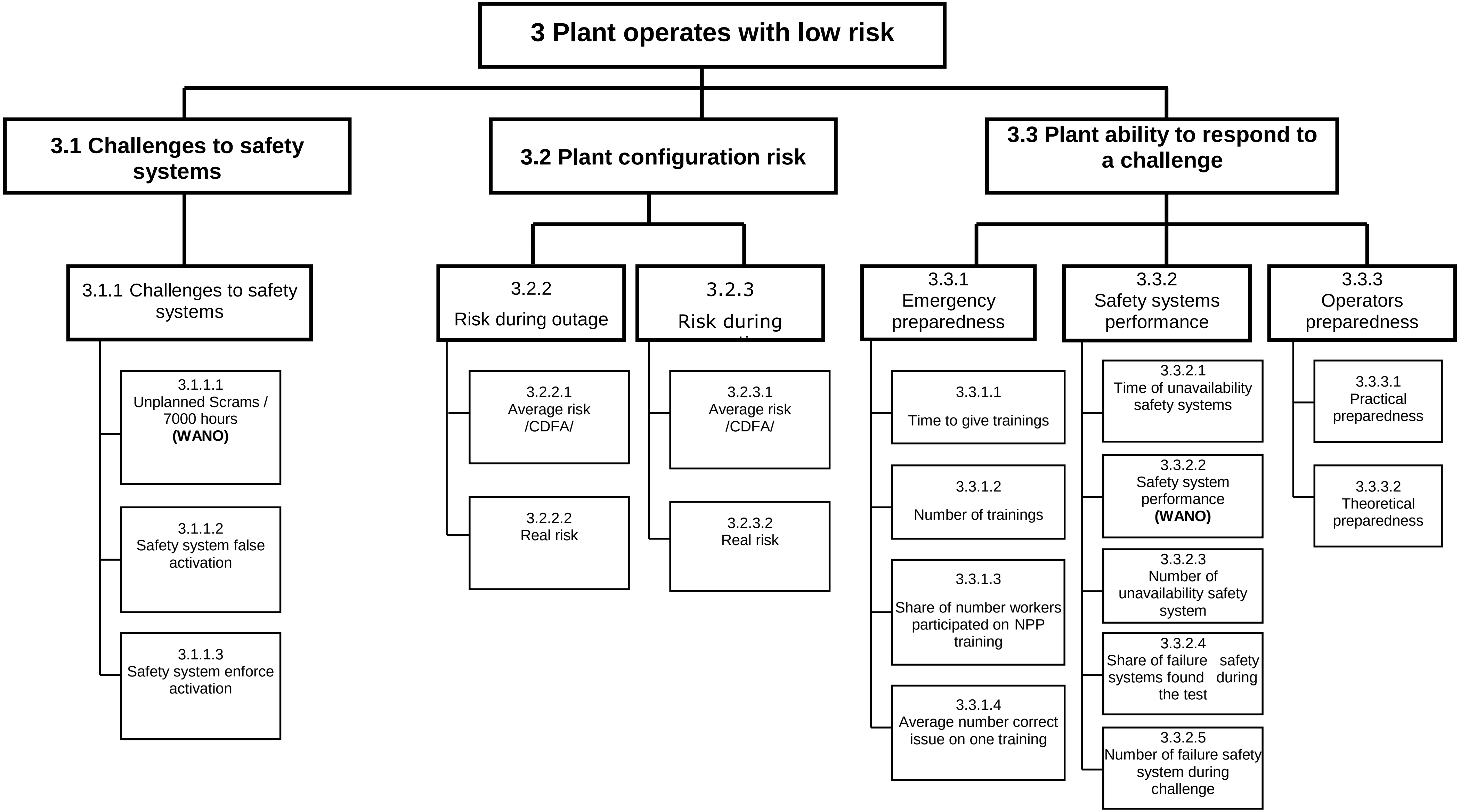
12. Data sponsor:	Name1: Nižnansky Marek - phone number : 2226	Name 2: Hacaj Augustin - phone number 2806	Name 3: Rosa Ľubomír - phone number 2577

13. Note:	WANO indicator	**14. In filled by:**	Name /centre: Nižnansky Marek -27062

SE structure of SPI



SE structure of SPI



18. Releases of iodine-131, noble gases and aerosols from Bohunice V-1 appear significantly higher than for V-2 and the Mochovce NPP. Is there a specific reason for this?

Response:
The higher activity of radioactive iodine in primary coolant of Unit1 was the reason of higher releases of radioactive iodine from Bohunice V1 NPP. The higher activity was caused by an untight fuel assembly but no untight fuel had been identified during the outage tests.

19. As gaseous emissions of the Bohunice plant are by factor of 2-3 higher than that of the similar Loviisa NPP and releases of PWRs in major Western countries: Is it planned to change the existing limits and to reduce the gaseous radioactive releases?

Response:
Primary limit for discharges is the exposure to member of critical group from both gaseous and liquid discharges. The activity concentrations are only secondary limits calculated from that primary limit. So the activity limits can vary for different countries or sites depending of the different conditions. The real values of the discharges represent less than 0,5% of annual limits for all items of discharges.

The limits for gaseous discharges have been established by the Public Health Authority of the Slovak Republic. The parameters of NPP design were considered and dose constraint for public exposure due to discharges (250 mikroSv per year for individual of critical group) established in the Regulation on Radiation Protection was a primary base for establishment of limits. The limits for the site of Jaslovske Bohunice will be probably changed after shutdown of units of NPP V1 in 2006 and 2008.
The operator has carried out during last years many important improvements of ventilation systems in NPP V1. The goal of this changes was reduction of effluents during outages.

20. The activity of noble gases, airborne particles, Tritium, C-14 (Bohunice NPP only) and iodine is monitored, but only annual releases of iodine, noble gases and aerosols are shown. Please clarify if data exists about releases of Tritium and C-14. If yes, could you present these figures?

Response:
Gaseous discharges of 3T and 14C are not limited at Bohunice NPP and their values for 2003 are:

Tritium (GBq)	943,873
14 C (GBq)	134,628

21. Is the off site emergency planning in Slovakia 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?

Response:
In Slovakia the off-site emergency plans are prepared by the county in close co-operation with the NPP operator. The operator is obliged by the law (Atomic Act) to provide for county, district, all necessary information, data. The off-site emergency planning uses results of the probabilistic safety assessment (PSA) to specify Emergency Planning Zones (EPZ) around NPPs in Slovakia and release categories. Detailed analyses are made for accident sequences, which have dominant contribution to the radioactive releases. The source terms, release categories, accident sequences and corresponding calculated numbers are available. Inn case of exercise or during real emergency all results would

be available for neighbouring countries for them to prepare the most effective response.

22. Apart from the notification of an accident as required by the Convention on Early notification will the Slovak emergency authorities and/or NPPs in Slovakia 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?

Response:
Detailed analyses are made for accident sequences, which have dominant contribution to the radioactive releases. The source terms, release categories, accident sequences and corresponding calculated numbers are available. Inn case of exercise or during real emergency all results would be available for neighbouring countries for them to prepare the most effective response.

23. Have terrorist attacks, like e.g. intended civil aircraft impact, been considered in safety analyses for the NPPs and which results have been yielded? Have measures been taken to minimize the risk of terrorist attacks against NPPs? To withstand the impact of which aircraft crash are the NPPs in the Slovak Republic designed?

Response:
In SARs, there is no aircraft crash assessed as a deliberate terrorist action.
An aircraft crash was considered to be an external event in SARs, which was assessed according SG 50-SG-S5 „External Man Induced Events in Relation to Nuclear Power Sitting", IAEA, Vienna 1981, with the following results. For both plants (V1, V2), the probability of aircraft impact for different kind of air traffic is $7{,}9.10^{-8}$ and lower, which is less then value of 1.10^{-7} recommended in safety guide. Therefore no protective actions against aircraft impact are required. Moreover, analysis did not include a restricted area around site (diameter 2 km, height of 1200 m), which would decrease calculated probability. Set of technical, regime and administration measures, which are included in security project of NPPs, have been taken for minimizing the risk of terrorist attacks against NPPs.

Belgium

1. Are there minimum requirements regarding the duration of re-training (e.g. number of days) or are there more detailed requirements on the content of the re-training programmes?

Response:
In 2004 minimum requirements are contained in the decree no.187/1999 and duration of two weeks were prescribed for re-training. Based on the new Atomic act the licensee will propose the program of re-training which will be approved by UJD.

Bulgaria

1. Could Slovak Republic present the target terms for implementation of the developed SAMG's for the different units?

Response:
The project for the SAMG development was the common project for both Mochovce and Bohunice NPP. This project was developed in cooperation with Westinghouse Energy Europe sprl. Brussels, Belgium. The project was initiated in January 2002 and finished in March 2004.
The schedule for implementation of hardware modifications may be affected by the privatisation of SE and changing legislative requirements applicable for severe accident management. Currently there are no legal requirements for implementation of SAMG in Slovak republic.

Current schedule for implementation of SAMG includes several consecutive steps, part of which has already been realised:

- Development of English version with WESE - completed 2004
- Review and development of Slovak version 2004/2005 – partially completed
- Verification and validation of SAMGs – stepwise process depending on the hardware modification development status – starting tentatively 2006
- Performance of analyses needed for technical specification of design modifications – 2005 underway
- Additional analyses of specific related problems – concurrently with design activities
- Training of NPP personnel in severe accident phenomenology – starting 2005
- Development of design modifications – starting 2006
- Training of TSC staff - 2007
- Installation of hardware modifications – tentatively until 2010
- Implementation only after the installation of modifications

2. The assessed dose for a member of the critical group of the population around NPP Bohunice for the last 6 years was between 0.08 μSv and 0.23 μSv/a. Which radionuclides have contributed mainly to those values?

Response:
The main contributor from liquid effluents is tritium and noble gases from gaseous discharges.

3. The report presents that "A project aimed to apply the in-vessel retention strategy using reactor pit flooding … is implemented … since early 2003". Some more information about this project would be appreciated.

Response:
The analytical project supporting implementation of the in-vessel strategy for Bohunice and Mochovce units was completed in 2003 – 2004. The original aim of this project was to apply the Loviisa NPP approach for VVER 213 units. However within this project we wanted to verify also the possibility to provide heat removal from the external surface of the reactor pressure vessel without lowering bottom heat shield, because the installation of necessary hydraulic systems would require major investments and may involve technical problems specific to the VVER 213 cavity. The conclusion of the project analyses is such that the heat removal can be accomplished without lowering the shield by water inflow through dedicated opening in the bottom of the shield and steam removal around the top of RPV. The analyses will be reviewed (possibly international mission) and the decision on the modification strategy will be taken afterwards.

4. The report presents the information that Level 1 SPSA in 2000 recommended to extend the operating procedures for normal and emergency operation for the shutdown unit to reduce the reactor core damage frequency and that the updated 2003 Level 1 SPSA ended with nearly the same result and recommendations. Can Slovak Republic explain in more details what was exactly implemented after the 2000 SPSA and what is the explanation on its small overall effect on the 2003 SPSA results?

Response:
In December 2003 symptom-based emergency operating procedures for full power of the reactor from Westinghouse were implemented into the V1 plant. They provide more adequate operating guidance to prevent the potential core damage through the optimal use of the safety systems. The results of the updated full power PSA study have shown us that the symptom-based emergency operating procedures reduce mainly the full power risk.
These procedures were developed for the full power operation of the plant and do not involve initiators postulated for low power and shutdown PSA (such as

loss of Reactor Heat Removal (RHR), loss of the spent fuel pool cooling, etc.). Therefore, they have only limited impact on the plant risk during shutdown.
The updated 2003 Level 1 SPSA study had to quantify this impact and overall effect on the results was small.
Recommendation to extend the symptom-based emergency operating procedures for low power and shutdown operation of the reactor is still valid.
It should be noted that the development of the extended symptom-based emergency operating procedures for low power and shutdown operation of the reactor was started in year 2003 and extended emergency procedures will be implemented into the V2 plant at the end of year 2005.

The SPSA results show that the contribution of human factor is dominating at the CDF. The Mochovce NPP initiated a corrective measure to improve procedures (Symptom Based Emergency Operating Procedures for a Shutdown States - SD EOPs) for shutdown reactor operation. The project is managed by BNFL on behalf of UK Department of Trade and Industry within the Nuclear Safety Programme. The development of the SB EOPs is performed with Westinghouse Electric Europe sprl. Brussels, Belgium collaboration. The project covers the development, validation and training material. The end of 2005 schedules the duration of the project.

Czech Republic

1. The improving results of PSA studies level 1 and 2 for all units in Slovakia are impressive, particularly for units V-1, which has to be shut down. Can you explain the big difference between PSA level 1 for full power and for SPSA for Mochovce units in comparison with Bohunice V-1 units. Why the results for SPSA for V-2 are not provided?

Response:
Differences between PSA studies of the Bohunice NPP and Mochovce NPP are due to different plant specific data and used plant specific initiating events frequency. Other reasons are slight differences in definition and determination of the plant operating states – POS and differences in the fault tree modelling approach. The Bohunice NPP and Mochovce NPP are in depended plants with slightly different operational and maintenance practice.
Main result of the Level 1 SPSA for Bohunice V-2 (CDF - Core Damage Frequency) is provided in the National Report on the page 15.
"The Level 1 SPSA results and conclusions following the implementation of certain Unit 3 upgrading measures in 2003 were CDF= 5.75E-05/year. "
Complex detailed results of the SPSA study is not possible to provide in this general report. The Bohunice V-2 NPP can provide this type of complex results on the specific request.

2. How the safety culture and motivation of plant personnel of NPP V-1 have been maintained under the politically motivated shut down in 2006 and 2008?

Response:
Following the Slovak Government resolution to early closure Bohunice NPP V-1 units in 2006/2008, the Action plan for ensuring high operational safety level for the remaining operational lifetime was developed in Slovenské elektrárne. According to this Action plan nuclear safety and operational reliability of the plant units is continuously reviewed and the Labour Agreement was agreed to maintain the Bohunice NPP V-1 staff motivation until Bohunice NPP V-1 final closure.

3. How the problems of liability of licensee for potential damages caused by radioactivity and by another aspects of nuclear installations operation are solved in Slovakia?

Response:
Concerning the nuclear damage and liability those are defined in the Act No. 541/2004 Coll II. The limitation of liability is laid down in § 29 Sec. 6 letter a)

and b) of Act No. 541/2004 Coll. II., and pursuant to § 30 Sec. 1 the licensee shall ensure that its liability for nuclear damage is covered by insurance or some other form of financial coverage for specified sums. As for the liability for another possible imminent damages from the nuclear installation other than nuclear damage are covered by the Civil Code and Commercial Code provisions.

4. Are there any plans to join the regulation of nuclear safety and radiation protection under one authority like in majority of countries? What are the major obstacles of this process?

Response:
Nowadays there is no plan to integrate nuclear safety supervision and radiation protection supervision into one and only administrative authority. UJD´s competences are laid down in the Act No. 541/2004 Coll. II. The competencies of the Ministry of Health Of Slovak Republic and of the respective Public Health Offices in the area of radiation protection are laid down in Act No. 272/1994 Coll. as altered and amended. Based on the resolution of Slovak Government No. 442/2003 from June 5th 2003 adopted as a result of an IRRT mission in November 2002, an agreement between UJD and Ministry of Health was signed on cooperation in performance of supervisory activities in order to avoid either duplicity or omitting some activities.

5. Which steps (legal, technical) have been taken for the future life extension (long term operation) of NPPs in Slovakia?

Response:
For example units 3 and 4 of Bohunice NPP have been and are being substantially upgraded by means of upgrading programs - NPP V2 Upgrade and Safety Improvement. Activities concerning application of future long-term operation as maintenance practices, equipment qualification, quality assurance, in-service inspections are in place. Final safety analysis reports are periodically updated after a period of 10 years and PSA studies are widely used. In many aspects the current situation in Slovak Republic is very similar to the existing situation in the Czech Republic.

6. Size of emergency planning zones of NPP Jaslovské Bohunice and NPP Mochovce are defined by Slovak legislation. Is there any consideration to make their revision for example with using results of PSA studies?

Response:
An effort has been made to take off fixed numbers of EPZ from relevant newly prepared laws and Regulations. The idea is to be more flexible in case of revision or change of EPZ. Naturally, in the case of a request to change EPZ, the operator of any nuclear installation has to submit an application or request supported by a set of documentation including PSA estimation proving that the change in question is justified and reasonable and would not result in the increase of risks connected with operation of given installation from the point of view of consequences.
The request together with all these documents is then to be reviewed by regulatory authorities, which can, if all conditions are kept on, to approve a new size of EPZ and issue relevant authorization for that.

Finland

1. Detailed and interesting information is provided on operating experience feedback in paragraph 5.3.5. What is the number of IRS reports written during the last eight-year period? Has analysis of events and feedback from the eight-year period shown any interesting technical or human/organizational factors from which other organizations, e.g. VVER users, can learn?

Response:

There are 16 Slovak IRS reports in the IRS database since year 1992, 11 reports from the year 1996 to 2003.

There is the strong evidence that the trend of the required events reported to national regulator at the Mochovce and Bohunice NPPs is decreasing. The relevant events are reported to WANO web site by the NPP. This system allows to each WANO member to use lesson learned from the Mochovce and Bohunice NPPs experiences.

2. Development of periodic safety review (PSR) practices is reported in paragraph 4.5.7 in 2001 and 2004 reports. Could you provide more information on the new (2003) regulatory requirement concerning content and periodicity of PSR? When will the systematic and comprehensive PSR take place for the first time?

   Response:

   The act No. 541 (Atomic Act) specifies:

   "(2) During the operation and during the decommissioning of a nuclear installation, the licensee shall be liable to perform regular, comprehensive and systemic assessments of nuclear safety (hereinafter referred to as "periodic nuclear safety assessments"), while taking into account of the current status of knowledge in the area of nuclear safety assessments, and to take measures to eliminate any deficiencies identified."

   Based on the former act No. 130/1998 (Atomic Act) UJD issued Regulation No. 121/2003 on Periodic safety review. The Regulation No. 121/2003 was prepared in accordance with the "IAEA safety guides PSR of Operational NPPs 50-SG-012".

   The UJD prepares a new version of the Regulation on Periodic safety review to follow requirements of the act No. 541/2004.

   Content of the PSR will be identical with the "IAEA safety guides PSR of Operational NPPs No. NS-G-2.10" Vienna 2003, but UJD or the licensee can extend its content. From the view of the UJD content of the "IAEA safety guides PSR of Operational NPPs No. NS-G-2.10" is understood as minimal content.

   Periodicity of the PSR is determined at 10 years.

   Periodic safety review of Bohunice V2 NPP after the second decade of operation is currently in progress and PSR final report have to be finished in 2006. Scope and content of this PSR Project is consistent with the UJD requirements given in the Regulation No 121/2003, which was in force in time of the PSR Project development. Also, recent recommendations in IAEA Safety Standard on Periodic Safety Review of NPPs were taken into account at the development of this Project.

3. Development of safety culture is reported in paragraph 4.3.3. Could you provide more information on safety culture indicators, self-assessments and action plans at NPP's, and their positive effects on the plant safety?

   Response:

   Safety culture indicators are involved in assessment of nuclear safety and operational reliability report. Safety culture indicators are: Share of human factor events, Injury factor, Number of Short-term Tech. specifications changes, Number of Violation of Tech. specifications, Share of Documentation shortcomings events, Number of deviations found through nuclear safety audits in which plant personnel did not follow procedures, Number of repetitive events. Safety culture indicators are assessed yearly in Nuclear safety committee.

   Action plans could be defined as a list of actions that assist in maintaining and improving the safety culture in NPP. These actions have responsibilities with dates of fulfilment assigned. The actions in Action plan for Bohunice NPP are divided into five fields - 1. Nuclear safety as a main priority in plant operation, 2. Improvement of feedback from operational practices, 3. Improvement of

personnel safety awareness and involvement, 4. Human factor events prevention, 5. Safety culture assessment.
The actions in Action plan for Mochovce NPP are divided into four fields - 1. Responsibilities and competencies, 2. Monitoring, assessment and analysing of safety culture, 3. Motivation and valuation of plant personnel, 4. Identified issues from last year.
Also the questionnaire survey was realized in Bohunice and Mochovce NPP. In the case of Bohunice NPP the first survey was carried out in 2002. The repetitive survey will be realized after 4 or 5 years. In the case of Mochovce NPP the survey was carried out in 1999 and a repetitive survey was realized in 2002. According to results from repetitive survey improvement in the safety culture is evident.

France

1. The report mentions (§2 p. 11-21 and §5.4 p. 84) the activities planned by the operator to improve safety. Could the Slovak Republic also provide information about the activities planned by the regulatory body for this improvement?

Response:
Safety enhancement programs of each NPP in the Slovakia were developed on the basis of plant specific safety review results. Safety review results including proposed measures for safety improvement are regularly reviewed, assessed and accepted by the Regulatory Authority.

Following are the activities of the regulatory body concerning safety improvement programs:
- Review and acceptance of the overall safety goals (strategy) of the Safety enhancement program
- Assessment and approval of all individual safety important modifications included in Safety enhancement program
- Approval of the time schedule for implementation of items or phases of the Safety enhancement program
- Inspections prior, during and after implementation of modifications

For example in case of ongoing NPP V-2 Upgrade and Safety Improvement the safety goals and the time schedule for implementation of safety improvement program phases were approved by the regulatory decisions No 214/200 and 250/2001. Individual modifications are currently reviewed and approved prior their implementation.

2. Could the Slovak Republic provide more information on the iodine distribution system (practical methods, difficulties encountered…)?

Response:
NPP operator provides for iodine prophylactics for own personnel and individuals who stay in its territory as well as for the public in NPP vicinity and for emergency staff and intervention units.
Iodine prophylactics are potassium iodide tablets containing 65 mg of KJ (per table) in packs of four. The manufacturer and supplies is VULM Modra.
District and regional offices´ civil protection units in the jeopardised areas specify the required numbers. Based on the data, the NPP operator orders and purchases the number of KJ packs needed.
Iodine prophylactics are distributed to towns and municipalities through civil protection components of the district and regional offices. The local and city offices distribute them directly to the public.
The district and regional emergency commission decides on the use of KJ following the recommendations from the NPP operator´ emergency commission.
No problems have occurred in the KJ distribution.

3. The report indicates (§ 4.7.2.1 p. 66), about the level 2 national emergency preparedness that the territory where measures are planned to protect the public are determined by a range of 30 km around the NPP Bohunice and of 20

km around the NPP Mochovce. Could the Slovak Republic indicate whether these differences are based on different hazard level or different criteria?

Response:
The emergency zones (30 km) around Bohunice site were designed and approved by the former CSKAE (Czechoslovak Atomic Energy Commission) experts for NPP A1 with HWGCR reactor before its commissioning. The conservative approach has been taken to determine the emergency zones. The distance of the zones were not changed neither for NPP V1 nor for NPP V2 during the construction, commissioning and commercial operation. Later CSKAE confirmed distance of emergency zones also for NPP V1 and NPP V2.
Later on, in eighties calculations were made by Nuclear Power Research Institute ("VUJE") using codes for severe accident consequences calculations available at that time taking into account design basis accident, relevant source term, core inventory, the worst meteorological conditions, etc. According to these calculations the distance relevant to calculation inputs would have been 17.6 km for NPP V-2. Thus it was confirmed that the suggested EPZ radius is sufficient one and the size of EPZ with radius 30 km was retained. It was declared that the radius 30 km is sufficient for preventive measures as well as in public protection.
The design of NPP Mochovce was made several years later than the one for Bohunice. New calculation of consequences, which could be caused by a potential severe accident, confirmed that the EPZ with radius 20 km is quite sufficient for this NPP.

4. Could Slovak Republic provide more information on the "complex list of operational safety performance indicators" used in the Power Plant Risk Control software and their collection?

Response:
The operational safety performance indicators (OSPI) system is used at SE, a. s., to assess the nuclear installation operational safety in accordance with the basic structure defined in Annex 3 and SE/MNA-051.01-01 - Operational Safety Performance Indicators.
The software PPRC - Self -assessment of operational safety for nuclear power plants - is used to gather and record data and work with OSPIs at IIS-SE.
The entire process of work with PPRC is described in the user's manual "Application of PPRC, release 1.83, for self-assessment of operational safety for nuclear power plants ".
The state of operational safety of SE-EBO and EMO units is periodically evaluated, at least on a quarterly basis by analysis of OSPIs and meeting the goals.
The assessment results are developed into the form of a "Report on the state of operational safety" (hereinafter referred to as the Report).
The Report is submitted for discussion to SE-EBO and SE-EMO management meeting and thereafter, once approved, sent to SE-HQ. It is also sent to the Nuclear Regulatory Authority of the Slovak Republic (ÚJD SR) in accordance with Act No. 541/2004 Coll.
SE, a. s. , HQ co-operates with special departments at the plants in preparing a complex annual report on nuclear installation operational safety. The software has been installed centrally at SE, HQ.

5. Could the Slovak Republic indicate whether the Regulatory Body has developed its own safety quality management system?

Response:
The Slovak Regulatory Body has developed its own quality management system, which is in force since January 1, 2002. The system is based on the STN EN ISO 9001: 2000, IAEA 50-C/SC-Q, and IAEA TECDOC–1090.
UJD starts to apply (first departmental evaluation was done 2003) also the EU public administration's Common Assessment Framework (CAF). CAF integrates the principles of the models from the European Foundation for Quality Management and from Speyer, the German University of Administrative

Sciences. Appointed team of 8 staff persons works to perform full internal audit and self-assessment for the whole UJD now. An implementation of CAF in Austria is managed by The Federal Chancellery, Dept. for Administrative Reform, Wollzeile 1-3, A - 1010 Wien, Tel.:+ 43-1-50190-7148.

6. Could the Slovak Republic provide information on regulatory actions to detect deficiencies in plant safety management?

Response:
Basic requirements on management of the nuclear installations are given in the Atomic Act (§7 conditions on licensee holder, §§17,18,19,20 requirements on the organization of NPPs, §24 requirements on personnel, §25 QA requirements).
According to national and international practice deficiencies in plant safety management can be characterized offer a reasonable delay. Regulatory actions to detect symptoms of deficiencies in plant safety management are based on the assessment of trend growth according to the results of:
- short-term (annual) assessment of the operational safety of NPPs (operation indicators, operating events, modifications, etc.),
- self assessment of operating organization,
- annual assessment of UJD´s inspection program results.

Regulatory body intervene on the regular meetings with the plant management in a case when corrective actions of operators are not appropriate.

7. Could the Slovak Republic provide more information on the updated Nuclear Safety & Radiation Protection Policy whose publication was scheduled in 2004?

Response:
The overall Safety Policy of SE, which specifies safety goals, requirements, policy, principles and responsibilities in individual areas of safety as nuclear safety, radiation protection, environmental safety and etc., together 14 areas.
Partial policies of individual areas will be linked to Safety Policy.
The Nuclear Safety and Radiation Protection Policy is ready for approval and will be issued together with other policies after approving by the board of directors of SE. This is expected in the first half this year.

8. The report provides (§ 3.1.5 p. 35-36 and also § 3.2.2 p. 37-38) with the various types and techniques for the NPP supervision by inspectors. It would be helpful to illustrate this information with some statistics regarding the number and the main topics of inspections performed during the current period (as is it only mentioned at the very end of § 3.2.2.1): this would facilitate the understanding of the main current safety issues. Could the Slovak Republic provide some information on this matter?

Response:
Statistics of inspection s performed by ÚJD SR inspectors in 2004 are given in the table below:

Subject of inspections	**Team insp.**	**Special insp.**	**Unplanned insp.**	**Routine insp**	**Summary**
SE – EBO	9	14	7	8	38
SE – EMO	4	13	3	4	24
SE – VYZ	2	11	1	4	18
VÚJE	0	1	0	0	1
PJM	0	4	4	0	8
KJM	0	33	10	0	43
Others inspections	0	2	0	0	2
Summary	15	78	25	16	134

Inspections in the table were carried out in the following areas:

QA – quality assurance	total number	6
TQ - technical qualification	total number	4
OP – operation	total number	29
MA – maintenance	total number	5
TS – technical support	total number	5
EP – emergency planning	total number	6
FP – fire protection	total number	12
PS – physical security	total number	8
RW – radioactive waste	total number	7
NF – nuclear fuel	total number	6
NFT - nuclear fuel transport	total number	7
NM – nuclear materials	total number	43
SAA – special air activities	total number	4
DD – decommissioning	total number	7

Note: Some of inspections were carried out in several areas.

9. The report indicates (§ 3.1.2.2 p. 27) that a new draft Atomic Act is currently prepared by the Government which will modify the Act 130/1998 and which will notably change the current system of permits related to nuclear installations as well as the position of the Regulatory body (UJD). Could the Slovak Republic provide information about the status of implementation of this new act as well as the mains characteristics related to nuclear safety and its supervision?

Response:
The new Act No. 541/2004 Coll. entered into force on December 1st 2004 except from provisions in § 3 Sec. 9 and 10 that shall enter into force on January 1st 2007. Concerning the most important changes brought by new Atomic Act in comparison with the previous Act No. 130/1998 Coll. II. are as follows:
Cancelling the issuance of the supplier authorisations for the nuclear energy industry by UJD that will not anymore regulate the supplier sphere. Such supplier authorisation was replaced by the permission granting for only specified activities. Thus UJD as a regulatory authority definitely has ceased from position of licensing authority for supplier sector as those reviewing duties and responsibilities of operator suppliers have been ceded to the operator themselves. UJD has focused the supervisory activities on the operators of nuclear installations and their quality assurance systems. The quality assurance system documentation has to be approved by UJD. The next important change is that of UJD became a specialised construction authority according to the Act No. 50/1976 Coll. the Construction Act. It means if there are nuclear installation constructions or constructions related to the nuclear installation, UJD became a final licensing authority for the stage of construction permission issuance as well as for the stage of construction authorisation procedure. According to the previous regulation, the construction authority for the nuclear installation construction was the Regional Administration Offices and UJD provided them with binding statements. In relation to the performance of supervisory activities, the position of inspectors towards the inspected persons, the enforcement procedures and sanctions imposed, the new Atomic Act did not brought significant changes in comparison with the previous Atomic Act.

10. The Slovak Republic has chosen not to address it its third report the aspects that remain unchanged from the previous report, which saves only 20% of the number of pages and obliges to review both document at the same time. In addition the fact that the regulation has not changed does not necessarily mean that there has been no progress in its implementation. In that sense a more self-standing report would have been appreciated which would have allowed to clearly highlight the trends, progress and difficulties in complying with the regulation.

Response:
The 2nd Review Meeting emphasised, that the next national reports should concentrate on new facts and changes and not to repeat facts contained in previous national reports. The rational behind this is that the attention of the reviewer is drawn on the most important changes. In addition the advantage of this approach is that questions are put only on the new chapters and questions on the old ones which were subject to previous review meeting are avoided. Slovakia made available on the internet all the previous national reports and answers to questions for new Contracting Parties. Because of the significant number of changes and the number of national reports Slovakia will review its approach for the 4st Review Meeting.

11. 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.

Response:
In its third national report Slovakia addressed all the articles of the CNS. For the convenience of the reviewer only new facts are contained in the national report, old ones are contained only in the amount and form to better understand the overall concept of the matter. For example it is not convenient to list old expert missions or the content of safety analyses reports (SAR) if they are overruled by new missions or SAR results.

Germany

1. Are there any difficulties in the new (including digital) I&C technologies in both hardware and software implementation from the point of view of their compatibility with existing ones?

Response:
No principal difficulties were found out in insuring compatibility of the new systems – safety systems are fully new with the minimum of interfaces with existing ones.
For assurancing the correct operation following measures were performed:
- measurement of electromagnetic compatibility for safety systems and especially for the neutron flux measurement,
- at the main control rooms the new systems were installed in separated boxes with new antistatic floor there.

In addition the following measures has been taken (e. g.):
- it is forbidden to use transmitters and portable telephones in safety system rooms,
- it is forbidden to make electric welding at the reactor hall and surrounding areas during operation of refuelling machine.

No issues are identified with regard to compatibility of new computer systems (I&C) with existing technological equipment and no operation events and operation constrains were identified as well.

2. What are the load assumptions for the containment design to withstand external hazards?

Response:
General requirements for the containment design (confinement in WWER-440 reactor type) are given in the article 22 of the Degree No. 167/2003 Coll. "On the Requirements for Nuclear Safety of Nuclear Installations".
Point (5) of article 22 in connection with the external hazards requires: "A containment system shall satisfy the requirements for protection against the external hazards referred to in Article 13".

In the article 13 are following requirements:

In designing, expect of the physical protection of nuclear installations and nuclear materials laid down by specific regulations, shall be taken into account:

- the most serious natural phenomena, historically recorded in the site area of nuclear installation and extrapolated having regard for the limited accuracy with respect to their magnitude and time of occurrence
- combinations of the effects of the phenomena triggered by natural conditions and human activity

A project shall design a nuclear installation protective zone to protect nuclear installations from external events which may be triggered by natural condition or human activity.

It means that the containment shall be designed for all justified load conditions.

3. Which acceptance criteria have been used for the regulatory review of the radiological consequences of design basis accidents? Are these criteria related to releases or related to radiological exposures? If dose limits are applied, which are the parameters (e.g. exposure pathways, integration times, distances) considered for the calculation?

Response:

The acceptance criteria used for regulatory review of radiological consequences are related to the radiological exposures and are established by the Ministry of Health's Regulation No. 12/2001 Coll. These criteria differ for different groups of workers and are different for exposures of citizens too. In a case of design basis accident, the radiological exposures of citizens are the same as for a normal operation. The Regulation also establishes the intervention limits for the immediate and consecutive measures, which may be relevant for the beyond design bases and severe accidents, only. The following parameters are considered in such calculations: – pathways - cloud, deposit and inhalation; – integration times - 48 h and 7 days; – distances - 3 km and 40 km.

4. What about the problem of HPSI pumps' reliable long-term post-accident operation in the sump suction mode for all power units regarding pump blocking by particle debris? Is the problem by the sump clogging in case of LOCA eliminated?

Response:

The problem regarding pump blocking by particle debris in the confinement was eliminated at all units in Bohunice NPP.

During gradual reconstruction of V-1 plant new modification was implemented for prevention blocking of ECCS pump suction.

In 1999 a 2000-year similar modification was implemented on units 3. and 4.

The safety injection sump suction clogging was examined within the safety improvement project as the Safety measure S05 before the start up period of the unit 1. The Safety measure S05 was divided in two parts: the first part focused on the analytical and experimental activities and the second part focused on the design activities. According to the results and conclusions decision was taken to perform the following modifications:

- Installation of new grid structures,
- New anchoring of grid structure installation,
- Implementation of two independent level measurements for each ECCS in front of grid structure and inside of grid structure, each measurement is working on different principle (conductivity, auxiliary pressure), measurement outputs are in each control room.

MAAE recommendations according to EBP-WWER-03 were satisfied.

Hungary

1. The events statistic shows, that more than half of the events was equipment failures, but the maintenance and safety improvement parts does not contain the answer to this problem.

   Response:
   Bohunice NPP analyses each event and undertakes corrective actions related to equipment modification including staff training. The programs run for failure sensitive equipment and instrument replacement.
   At the Mochovce NPP the effectiveness of corrective measures to address equipment failures is evaluated regularly. The indicator of repeated events is a part of this evaluation. Trend analysis is used to take appropriate corrective measures for equipment maintenance as well as equipment modifications. The trend of the events relating to the equipment failures is decreasing.

2. Please clarify if the safety improvements are based on, deterministic or PSA studies.

   Response:
   Safety improvement measures included in the safety upgrading programs of all NPPs in the Slovakia are based on outputs of the deterministic safety reviews of the original plant design. Results of the IAEA generic safety review for WWER 440/V230 and WWER 440/V213 type reactors supplemented with the results of national safety review of NPP design and/or with the results of additional international safety review (PHARE Program) were used as a basis for the development of plant specific safety upgrading programs.
   Probabilistic PSA studies developed at that time were used for prioritization of safety important safety measures in the implementation time schedule as well as for the assessment of the safety level reached after the implementation of upgrading programs. However, most important outputs from PSA were additionally incorporated in the safety upgrading programs (e.g. implementation of symptom-based operating procedures).

3. In 2001, UJD approved the decision "Safety Concept for NPP V-2 Upgrade and Safety Improvement". The chapter does not contain information about the execution.

   Response:
   The basic design and particular tasks ("Modernization Tasks") within NPP V2 Upgrade and Safety Improvement (Modernization) based on "Safety Concept for NPP V2 Upgrade and Safety Improvement" have been developed and is being developed. Upgrade and Safety Improvement process contains about 90 measures covered by particular tasks. Performance of individual Modernization Tasks has been scheduled for the period of 2002 ÷ 2008. It overshoots the period according to the actual approved time schedule.

Japan

1. Reference: 5.3 Operation, 5.3.5.2 Documentation and analysis of events at nuclear installations, Events without consequences (near miss), on page 79 and 80;
   A system of reporting and feedback from minor events was implemented in 2000 as a measure intended to improve the safety culture. … A staff is encouraged by the plant management to report minor events. Reports on minor events are registered and evaluated by the Feedback Group. … SE plans to implement in 2004 a project in co-operation with United Kingdom to evaluate and improve the efficacy of the current system of handling near misses.

Huge amount of minor events, from which operators would be able to extract lessons learnt, are unreported in the world. The above-mentioned system should contribute to the nuclear safety improvement. The results of the implementation of the system and the project should be very much interesting for all the Member Countries in the IAEA and therefore should be appreciated to be reported.

Q1: Does the report of near miss include human and organization relevant events? If so, please explain human and organization-related improvement in the past based on the report.
Q2: When the co-operative project with UK is planned to complete? The UJD is going to include the summary of the results into the national report for next nuclear safety convention?

Response:
Q1: The human and organizational failures are also reported within the near misses system. The main goal is to take corrective measures for individual near miss reports and collect that important information, which could help to the NPP organization to learn and improve safety culture. The safety culture is evaluated at the Mochovce NPP and Bohunice NPP regularly. The individual corrective actions are implemented with the aim to reduce human factor failure or to improve organizational processes. One of the results of near misses reports is the evidence of decreasing trend for events with consequences.

Q2: The project for the improvement of near misses reporting was initiated in 2004. All of the aspects of technical failure, human failure and organizational failure are evaluated within the project. The project is not finished yet.

2. Reference: 4 General safety aspects, 4.5 Safety assessment and verification the 2nd para. from the bottom of 4.5.3 Basic principles of UJD issued decisions on safety improvements of NPPs in operation… another principle employed by UJD… is to limit the duration of operation of NPP units by giving permission for a limited period of time, which allows to manage the safety measure implementation process.
Q1: How is the length of the period decided? Please explain briefly the process to fix the limited period. Is the plant involved in the decision-making?
Q2: In case that the plant did not finish the safety implementation by the limited time due to e.g. some technical reasons, how does UJD regulate the plant?

Response:
The regulatory principle to limit duration of the operational license was introduced during the upgrading of Bohunice V1 NPP with WWER 440/V230 reactor type units. Safety Upgrading Project was distributed into several stages.
Q1: Before completion of the NPP V-1 upgrading program regulatory permit to continue with plant operation was issued for one fuel cycle prior the beginning of the fuel cycle. An application for this permit consisted of:
- safety documentation (assessment of operational safety during the preceding fuel cycle, fuel loading pattern, equipment ageing assessment, ISI results including annual reassessment of RPV status, etc)
- confirmation of implementation of the annual safety upgrading program (annually approved by the regulatory body)
- progress report on the implementation of the overall safety upgrading program (demonstrating achievement of the safety goals of the entire safety enhancement program)
- proposal of the detailed work program for the next period

Involvement of the plant is given above.

Q2: In the case with acceptable justification when the plant was not able to finish the safety implementation regulatory body regularly accepted exception in the annual program, but no exceptions were accepted to overall goals

3. Reference:4 General safety aspects, 4.5 Safety assessment and verification. The last para. of 4.5.3 UJD has set the probabilistic targets for acceptability at systematic level of safety systems, for the reactor protection system, for core damage, …

Q1: Regarding the probabilistic targets set by UJD, more explanation is appreciated. For example, what does "the probabilistic target for the reactor protection system" mean? What does the target value represent in this case?

Response:
"Probabilistic safety goal for the reactor protection system" is the value of the probability of the RPS system failure per demand. This value is calculated using probabilistic methodology.

4. Reference: 2.3 Nuclear power plant Mochovce – Units 1 and 2 , 2.3.3.3 Implementation of safety measures Completion of post-emergency monitoring means was carried out, having regard for elaboration of severe accident management guidelines (SAMG's).
It should be appreciated from the view point of radiation protection that Slovakia is enhancing safety monitoring aspects introducing such as post-emergency monitoring (2.3.3.3), a new monitoring program and system (2.5.3.3 on page 22 and 4.2.2 on page 44, respectively) for ISFSF and a teledosimetric system set the vicinity of NPP Bohunice (4.6.3.3 on page 62).
Q1: What do you intend to monitor as for post-emergency monitoring? How long the post-emergency monitoring is expected to perform after an accident?

Response:
For post-emergency monitoring, radiation monitoring points were specified. Basis for this was **the US NRC Regulatory Guide 1.97 rev. 3**. „Instrumentation for light-water-cooled nuclear power plants to assess plant and environs conditions during and following an accident".
Requirements for design and qualification criteria (redundancy, power source, channel availability, QA, display and recording, range, equipment identification,) for the radiation measurement instrumentation were also taken from the RG.
The points were: primary containment area radiation, containment effluent radioactivity from identified release points: stack, primary circuit-secondary circuit-BRUA valves, reactor hall, cooling water of the ECCS system.
Above requirements of the GR 1.97 rev.3 we realised teledosimetry (TDS) system in two measuring circuits. The first is just around reactor buildings (dose rate measurement from background value up to 10 Sv/h in lead shielding and decontaminable cover). The second circuit consists of air conditioning cabinets with dose rate measurements and aerosol and iodine sampling systems in each of the 16 sectors around the NPP. The distance of cabinets from the NPP is between3-8 km. Moreover in the three towns with higher number of inhabitants there are installed cabinets with dose rate measurements and continuous aerosol and iodine measurement. There is available specific software for release calculation, based on the TDS system measurement.
The devices have to be used according to requirement of the RG 1.97 rev.3. It means, it should be used until the source of radioactivity is under control of the operator.

5. Reference: 2.5.1 Description of the technology used The storage facility was upgraded to enhance its storage capacity, extend its service life time and upgrade its seismic resistance.
Q1: Were the local residents involved in the decision making process of the storage capacity extension and service life extension?
Q2: The storage capacity is going to be tripled. How long is the service life time extended?
Q3: A license was required for this modification? Please explain briefly a regulation process for modification of a nuclear installation.

Response:
Q1: The local public were involved in the authorisation process by discussing the Report on the assessment of environmental impacts of the interim spent fuel storage facility reconstruction under Act No. 127/1994 Coll., and the municipality of Jaslovské Bohunice was involved during the building permit procedure. The public discussion of the Report on environmental impact assessment called by the municipality concerned was attended by the representatives of the designer, supplier authorities, the Slovak Ministry of Economy, ÚJD SR, local residents, other public and NGOs.
Upon acceptance of their requirements and comments, all those present consented to the investment intent.

Q2: A minimum service life of 50 years (Safety Documentation, Chapter 3) is planned. ÚJD SR has given thus far permission on operation of the upgraded interim spent fuel storage facility until 31 December 2010. As of this date, SE-VYZ will evaluate the overall state of interim spent fuel storage facility and technologic parts and systems and apply for permit on operation thereof for further period.

Q3: For the modification of the Interim spent fuel storage facility a license from Building Office was required. To this license a standpoint was given by the NRA SR. After finishing modification a permit for usage was issued by the Building Office and an operating license was issued by the NRA SR (Decision Nr. 152/2000).
Please explain briefly a regulation process for modification of a nuclear installation.
Before 1 December 2004:
Regulation process was equal to previous case. License for modification was issued by the Building Office (a statement by the NRA SR was required), a permit for usage was issued by the Building Office. Operating license was issued by the NRA SR.
After 1 December 2004:
In 1 December 2004 a new Atomic law entered into force. According to this law all licenses are issued by the NRA: license for modification, permit for usage and operating license.

Lithuania

1. Which two types of mandatory documents do you mean?
The documents subject to approval include a radiation protection assurance quality program. What is that? May be it is a radiation protection plan? (3.1.4.1)

Response:
Radiation protection assurance quality program is a part of radiation protection program.

2. Please provide details, how the activities between the UJD and Slovak Public Health Office are co-ordinated among each other. Are there any agreements established? (section 3.1.1)

Response:
The most recent agreement on co-operation between UJD and Ministry of Health was signed on 23 June 2003 for five years period with the aim to assure effective co-operation of inspection activities among each other. In the frame of this agreement there was also established a Joint Committee for solving common issues, respectively for organising common meetings at least two times per year and on case-by-case basis. In the Statute to this agreement, there are specific items that define our co-operation.

3. Could you explain why quality assurance programme set up in accordance with outdated versions of ISO standards? (4.2.3)

   Response:
   The QA documentation for training and education of the staff and third persons is prepared and maintained in accordance with STN EN ISO 9001:2001 - Quality Management Systems - Requirements (it was mistake at the Reports), but in the area of Environmental Management Systems is the training documentation prepared really on the base of STN EN ISO 14001:1996 and nowadays are implemented in the documentations the requirements on the base of STN EN ISO 14001:2004.

4. Are there safety culture action plans and self-assessment parts covered in QA systems at NPPs? (4.3.3)

   Response:
   Safety culture action plans are issued yearly as a Plant Director Order. Plant Director Orders are defined as a Operative Management Documentation and this is involved in QA management system documentation. Accordingly safety culture action plans are covered in QA systems at NPP.

5. Which safety culture indicators are being used and how they are measured? (4.3.3)

   Response:
   There is used different way between Bohunice NPP and Mochovce NPP. In Bohunice NPP the safety culture indicators are as a part of overall plant assessment of nuclear safety and operational reliability. There are these safety culture indicators: Share of human factor events, Injury factor, Number of Short-term Tech. specifications changes, Number of Violation of Tech. specifications, Share of Documentation shortcomings events, Number of deviations found through nuclear safety audits in which plant personnel did not follow procedures, Number of repetitive events.
   In Mochovce NPP there are 40 safety culture indicators being monitored. Some of them are at once also operation indicators. Indicators are divided into four fields - Operational Safety, Radiation Protection, Qualification and training, Review and assessment (see also Finland 3).

6. Why does the Regulation lay down requirements for too many quality systems in the list? Sometimes it may be a headache for license holders. (4.4.1)

   Response:
   The structure of quality system was given by Act No. 130/1998 Coll. on peaceful uses of nuclear energy and UJD Regulation No. 317/2002 on the requirements for quality systems of licence holders); These requirements follow the requirements of IAEA code and guidelines 50-C/SG-Q (comment: the act No. 130/1998 Coll. was replaced by Act No. 541/2004 Coll.).

7. NIP inspection activity is described in 4.4.5 “Role of regulator”. NIP is not in the list of abbreviations. What is that? (4.4.5)

Response:
NIP is the abbreviation for the National Labor Inspectorate (industrial safety).

8. Concerning “Staff dose and exposure limits are established for quarter and annual periods”. What is the difference between staff doses and exposures? For what purposes quarterly limits do serve? (section 4.6.2)

   Response:
   Annual limits for doses arising from the exposure to ionizing radiation are set by the national legislation on Public Health Protection. This legislation does not says anything about the difference between “staff doses and exposure limits”. Quarterly limits are internal limits set by the operator of NPP.
   Probably there is a mistake in the translation. The correct term is dose limits. Dose limits for workers and public are established by the Regulation Ministry of Health of the Slovak Republic On the requirements for securing of the provision of Radiation Protection . There are annual limits and five consecutive years limit of effective dose. There are no quarterly limits. The operator can use operational dose constraints or reference levels for dose regulation purposes. These values are not limits.

9. Would you please clear up the concept of the “individual dose equivalent”? (section 4.6.3.3)

   Response:
   The limitation and concept of individual dose equivalent is based on legislation of Slovak Ministry of health (Act No. 272/1994 on protection of health and degree No. 12/2001 requirement to assurance of radiation protection). Both documents are based on the IAEA and ICRP standards and they are elaborated into the QA programme in NPPs.
   Annual individual limit of inhabitants is 250 microSv/year. In the practices the IDE value is estimated and based on the model calculation of real radioactive releases from NPP. The input data are a real release, metrological, geographic and demographic situation and output is IDE.

10. As we understood from the Report, a few organizations and institutions in Slovak Republic (Laboratory of Environmental Radiation Control, Slovak Radiation Monitoring Network Centre, Centre for Protection against Radiation, Slovak Public Health Authority) are involved in Environmental radiation impacts monitoring and assessment. Could you explain, please, how the coordination and exchange of information between these organizations is arranged and how UJP (Nuclear Regulatory Authority of the Slovak Republic) is involved in this process? (section 4.6.3.3)

    Response:
    There are five monitoring networks on the Slovak territory belonging to following ministries: Ministry of Environment, Ministry of Interior, Ministry of Health, Slovak Army and Ministry of Economy (NPP EPZ local networks). The network of Ministry of Environment is on-line connected to Emergency Respond Centre (ERC). In case of emergency data from all these networks are sent to the Slovak Central Service for Radiation Monitoring. Here data are collected, registered, recorded and evaluated. The results of evaluation of radiological situation are then available for decision maker on the national level and in the ERC of UJD.
    Note: A special permanent group of experts nominated by National Emergency Commission for Radiation Accidents is present in the ERC of UJD during the emergency. All results concerning, in general, the assessment of facility technical status, source term, technical, meteorological and radiological parameters are available for this group to be able to make qualified recommendations and materials for decision making.

Poland

1. How many inspectors are performing inspections and how many inspections is made in one unit per year?

   Response:
   33 inspectors from UJD SR conducted inspections in the nuclear facilities in Slovakia in 2004.

   Overview of the inspections conducted on all NPP units are given in the table below

			insp.number
NPP V-1	unit 1	EBO 1	**22**
	unit 2	EBO 2	**23**
NPP V-2	unit 3	EBO 3	**25**
	unit 4	EBO 4	**24**
NPP EMO	unit 1	EMO 1	**20**
	unit 2	EMO 2	**21**
	unit 3,4	EMO 3,4 (under construction)	**1**

2. What are the plans for construction of units 3 and 4 at Mochovce NPP? From experience with units 1 and 2, what kind of changes in design could be anticipated for subsequent two units if any?

   Response:
   Answer to the first part of the question:
   The project crucial time framework is the start-up of operation of Unit 3 and of Unit 4 by 31 December 2010 and 31 December 2011, respectively. The above dates are real provided that the activities of the project pre-preparatory and preparatory phases are fulfilled in material and temporal terms.
   The draft project for completion of NPP Mochovce Units 3 and 4 has been phased in material and temporal terms as follows:
   Phase 1: pre-preparatory by 31 December 2004
   Phase 2: preparatory from 1 January 2005 to 31 December 2005
   Phase 3: implementation from 1 January 2006 to 31 December 2011

   Under the pre-preparatory phase, the core activities defined in the Strategic plan for conservation and protective works on Units 3 and 4, including the performance of conservation and protective works, complex re-assessment of the quality of selected equipment and selected building constructions, were carried out.
   Other crucial activities under this phase were the preparation of the Terms of Reference for the change project and Phase 1 of the technical and safety concept, as-built documentation of the building part with incorporation of built-in technology in electronic format, and the elaboration of NPP Mochovce Units 3 and 4 completion cost update. Under the investor internal activity, a model was developed for project management and funding the completion of NPP Mochovce Units 3 and 4 and a revision of the Strategic Plan for 2005-2007. As regards legislation, change in the building permit was ensured, whereby the deadline for structure completion by 31 December 2011 was established. All the scheduled activities under the pre-preparatory phase in 2004 were fulfilled.
   Activities have been scheduled for the 2005 preparatory phase including a revision to safety documentation and quality documentation, the drawing up of contractor selection documentation, of appendices to the Basic Design, advance implementation activities.

   Answer to the second part of the question:

According to the original intents, Mochovce Units 3 and 4 were erected as flow construction following Units 1 and 2, which were commissioned in 1998 and 2000, respectively. Merely works to prevent degradation of the already implemented work have since been carried out on Mochovce Units 3 and 4. The IAEA recommendations contained in the advisory document TEC-DOC 1110 - Management of Delayed Investment Projects of Nuclear Power Plants - are followed. Under the preparation for completion of the half-finished Mochovce Units 3 and 4, background documents thereon to the extent of the changes have been drawn up with the aim of:
enhancing nuclear and operational safety up to the currently required standards,
accepting technological progress on nuclear energy, improving technical and economic parameters, complying with ÚJD SR requirements for NPP completion and operation in accordance with the current legislation.
In line with the effective Atomic Act, a document titled ”Terms of Reference for Change Project MO34” was prepared in 2004, incorporating safety measures from EMO1,2, having regard for certain solutions to upgrade of EBO and other power plants.
Types of planned changes
a.) Safety problems to the extent of design accidents
Safety problems relating to the design of EMO1,2 are included based on their safety relevance into four categories. The categorisation of problems follows the IAEA-applied approach to VVER 440/213 in IAEA-EBP-WWER-03.

The respective problem categories groups were structured and designated as follows:

Code	Description	Number of measures
G	General	3
RC	Reactor core	1
CI	Component integrity	6
S	Systems (machine)	17
SKR	Control and management	11
OP	Dosimetry systems	9
EL	Electric systems	7
KONT	Buildings and constructions	6
IH	Internal hazards	8
EH	External hazards	3
AA	Accident analyses	15
TH	Above-design accidents	5

b.) Innovation of systems and equipment
Replacement of Automated Technologic Process Control Systems (I&C)
Modification to radiation control systems
Innovation of power transmission, self-consumption and electric equipment power supply
Request for unit power transmission
Neutron noise diagnostics system (in-core and ex-core)
Main circulating pumps
Storage pool cooling system innovation
Principal innovation of turbo generators and secondary circuit modification
Process water system innovation
Loading machine innovation
c.) Improving technical and economic parameters
Getting units involved in both the primary and secondary regulation
Boosting electricity generation
Fuel cycle innovation
Service life extension to at least 40 years
e.) Experience with EMO12 feedback
Changes adopted as outputs from the feedback process from operational events, where operational events and near-events are evaluated, their analyses and transfer of the analysis results back to the designs. This process also involves events from other NPPs.

Spain

1. Why are access controls not established among the protective measures to be implemented in the event of emergency situations?

   Response:
   The protective measures are applied immediately for NPP workers as well for other persons present at site after declaring the event at the nuclear installation that was classified as 2nd or 3rd level. Following immediate protective measures are performed after warning and notification:
   - Accounting and personnel movement control at NPP site
   - Assembling and sheltering
   - Iodine prevention
   - Personnel individual protection
   - Evacuation of persons from NPP site

   <u>Accounting and personnel movement control at NPP site:</u>
   All persons are subject to accounting and personnel movement control at NPP site
   Methods:
   a) Electronically (AKOBOJE)
   Electronically accounting and movement control is done by part of the security system „AKOBOJE". With the technical means the AKOBOJE system is able to provide the number of personnel present not only at site but also in internal areas of NPP.
   b) Physically – manual count of persons in assembly points and in shelters.
   All persons are gathered in assembly areas or in shelters after declaring of 2nd or 3rd level of nuclear event. The accounting of persons in shelters or assemblies is realized electronically by AKOBOJE system or manually by members of shelter or assembly group.
   c) Managers have knowledge about their subordinates
   Managers must have knowledge about the movement of their subordinates. Workers are obliged to notify their bosses when they are about to leave the working place and to determine the place where they will stay.
   When the EMERGENCY is declare the access control to NPP site is performed as well. In that case the main NPP gates are closed and enter or exit from the NPP is managed according to the NPP Emergency Plan. AKOBOJE system is used to manage it.

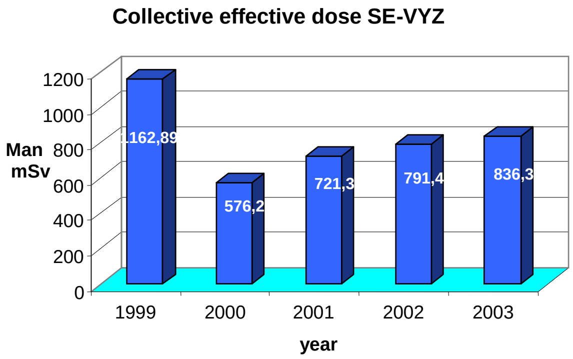
2. The rules applicable to any type of inspection include the inspection protocol, which indicates that the corrective measures established for the problems encountered should be recorded therein. Is it to be understood that the inspector is also responsible for establishing these corrective measures? If this is not the case, who is responsible for establishing these measures?

   Response:
   Each of inspectors from inspection team can require appropriate corrective measures. Responsibility to prepare list of corrective measures have all members of the inspection team. Head of the inspection team determines final content of all corrective measures included in the protocol.

3. Information is provided only for plants. (Comments)

   Response:
   Fig. 4.6.3.1a shows average collective dose developments at SE VYZ for 1999 through 2003. The 1999 collective dose is relatively higher because of the preparation and transport of spent fuel from the NPP A1 to the Russian Federation. There has been a visible increase in CED since 2001 due to the gradually growing amount of work on the project on decommissioning and commissioning novelty radioactive waste processing technologies.

CED diagram (Fig. 4.6.3.1a)



Tab

Atmospheric discharges in 2003			
Installation	Type of discharge	Activity	Share of site-wide limit [%]
Interim spent fuel storage facility	rare gases	---	---
	aerosols	0.648 MBq	0.0004
	iodine 131	---	---
Bohunice Processing Centre	Rare gases	---	---
	aerosols	5,036 MBq	0,0032
	iodine 131	---	---

Hydrospheric discharges in			
Installatio n	Type of discharge	Activity	Share of site-wide limit [%]
VYZ	corrosive and fissile products	85.296 MBq	0.23
	tritium	2,411.095 GBq	5.17

4. Section 4.7.6 indicates the systematic approach used for the performance of drills at specific installations, but not what is established in the applicable regulations. With what frequency are the on and off-site emergencies plans drills and evacuation simulations performed and what are the basic requirements applied? Why are the radii of the areas considered to entail a risk used in emergency planning at Bohunice NPP (20 Km) and Mochovce NPP (30 Km) different?

Response:
Drills and exercises
According to the Slovak legislation the emergency preparedness needs to be ensured by drills and exercises. The emergency drills and exercises schedule containing their kinds and extents is issued for each year. The NPP submits the schedule to regulatory bodies – ÚJD SR and Ministry of health.
In that manner the member of shifts, the emergency groups consisting of the non-shifts workers as well as the persons not involved in emergency response are trained.
Whole site exercise is performed once a year and every 3rd year there is the exercise performed with the cooperation of the off site – according to the legal requirements of Slovak republic. The drills of shifts involved in emergency response are performed at least two times a year.
Specific drills and exercises are run in specific areas of NPP. They are focused for example in the evacuation of the persons from the buildings, fire-fighting actions, security actions, etc. The specific drills and exercises

are performed according to the approved schedule once a month also out of working time. Besides the scheduled drills and exercises there also performed non-planned drills and exercises that are done according to the operational needs or regulatory requirements.

Emergency zones

With the reference to the Law 42/1994 and corresponding Regulations the emergency zone for EBO NPP has the radius 30 km and for EMO NPP 20 km.

SE EBO site is the common site for 3 NPPs (A-1, V-1, V-2). Originally the emergency zone with the radius of 30 km was established for the A-1 NPP. During the construction of further NPPs V-1 and V-2 the existing emergency zone was considered to be sufficient. In order to review the emergency zone the analysis is elaborated at present involving the new conditions (source terms for V-1and V-2). As the result the change of present emergency zone radius is expected

5. This section mentions that activities for the implementation of severe accident management (SAMG) are under way. In implementing these activities, is consideration given to the need to develop severe accident management guidelines? If so, will techniques be applied for the verification and validation of these guidelines? Is a simulator used to perform this validation in real time or is such validation based on previously simulated scenarios and graphs?

Response:

The first phase of SAMG implementation process – the development of SAMG – Severe accident management guidelines - has been completed in 2004. The guidelines have been developed and optimised for the NPP status after realisation of a set of hardware modifications in several areas, the major related to hydrogen management and in-vessel retention strategy implementation. Technique applied for validation of SAMGs (recalculated analyses or plant simulator for SAs) will depend on the available tools as of tentatively 2006 – 2007. So far no decisions have been taken.

An English version of the guidelines is available now. Activity on their translation to Slovak language is under way, completion of it is expected by the end of 2005. The strategy of the SAMG is based on a successful flooding of the reactor cavity and cooling of the reactor pressure vessel from outside, preventing thus a breach of the vessel. Although the existing analyses and experience of the foreign NPPs (e.g. Loviisa NPP) have been considered in strategy development, new analyses for support and motivation of the strategy will be required, too. A process of definition of the boundary and initial conditions has been initiated. It will lead to selection of the accident scenarios necessary for further analyses. However, a complete implementation of SAMGs will require a period of at least 2-3 years. Therefore not all the details of the process has been already defined, agreed and harmonised.

6. This paragraph states that in the event of degradation of any area of safety corrective actions are performed in order to prevent further degradation. A more detailed description would help to better understand this paragraph, and especially the activities that the regulatory authority carries out in this process.

Response:

According to the national and international practice deficiencies in plant safety management can be characterized offer a reasonable time delay. Regulatory actions to detect symptoms of deficiencies in plant safety management are based on the assessment of trend growth according to the results of:

- short-term (annual) assessment of the operational safety of NPPs (operation indicators, operating events, modifications, etc.)
- self assessment of operating organization

- annual assessment of the Regulatory Authority inspection program results

Regulatory body intervene on the regular meetings with the plant management in a case when corrective actions of operators are not appropriate.

7. This paragraph points out that safety improvement programmes are undertaken by the licensee of the installation, who assumes overall responsibility in this respect. Are these improvement programmes carried out periodically? How does the regulatory authority intervene in this process? Are there any criteria in place to determine in what cases approval of the improvements by the regulatory authority is necessary?

Response:

Safety enhancement program of each NPP were developed on the basis of plant specific safety review results (i.e. PSR or particular design safety review required by the regulatory body). Safety review results including proposed measures for safety improvement are regularly reviewed, assessed and accepted by the Regulatory Authority.

Following activities concerning safety improvement programs are subject to regulatory actions:

- Review and acceptance of the overall safety goals (strategy) of the safety enhancement program
- Assessment and approval of all individual safety important modifications included in safety enhancement program
- Approval of the time schedule for implementation of items or phases of the safety enhancement program
- Inspections prior, during and after implementation of modifications

United States of America

1. What is the specific authority of the regulator to direct a plant shutdown if they identify a condition adverse to safety at one of the nuclear installations? What criteria are used to make this decision?

Response:

Following criteria are used to make this decision according to § 32 of Atomic Act (No 541/2004):

The Authority shall decide to restrict the scope or the validity of authorisation shall order the authorisation holder to take the necessary measures or shall rule to suspend the operation of a nuclear installation where there is risk in delay of or upon a serious occurrence of nuclear safety, physical protection or emergency preparedness relevance.

2. What models were used for determining human error probabilities for each plant's Probabilistic Safety Assessment (PSA)? Have these models been validated against operating experience in Slovakia?

Response:

The human contribution to risk and safety of J. Bohunice V1 NPP is assessed and quantified using standard techniques of the human reliability analysis.

The Human Reliability Analyses for the J. Bohunice NPP is based primarily on THERP (Technique for Human Error Rate Prediction [Swain, A.D. and Guttman, H.E.: Handbook of HRA with Emphasis on NPP Applications, NUREG/CR-1278, August 1983]) and TRC (Time Reliability Correlation) [Dougherty, E.M. and Fragola, J.R.: HRA: A Systems Engineering Approach with NPP Applications, John Wiley & Sons, NY 1988] methodology - consistent with the Systematic Human Action Reliability Procedure (SHARP)

In the SPSA model the operator responses after an initiating event are evaluated using the success likelihood index methodology (SLIM) that is specifically developed to provide consistent assessments and detailed

documentation of these dynamic activities. Human error probabilities are received as result of the analyses.
In addition, the identification of dependent operator actions was performed. Preceding operator actions that have occurred before the required response can have very strong influence on the human error probability. Therefore, for some human errors the dependent and independent human error probabilities were calculated.
The Human Reliability Analyses and models have not been validated against operating experience in Bohunice NPP due to lack of real data of the human error events.

Company SAIC (USA) expert evaluated the HRA for the Level 1 PSA model at the Mochovce NPP. The quantification of HF was performed with THERP and SAIC TRC methods. The expert from UJV Řež organization performed the HRA for SPSA. The quantification was performed with THERP and SLIM methods.