ANSWERS TO QUESTIONS
ON
NATIONAL REPORT OF
THE SLOVAK REPUBLIC

COMPILED ACCORDING TO THE TERMS OF THE CONVENTION ON NUCLEAR SAFETY

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APRIL 1999

GENERAL

In the text of the Report there are no references to the Article of the Convention which the given part of the text (chapter, section, paragraph) belongs to, and the Table, given in page 9, on correspondence between sections of the Report and Articles of the Convention is incorrect almost completely. There are no conclusions in the Report on meeting the requirements of the Convention both on the whole, and by separate Articles. All this makes the reading and assessment of the Report extremely difficult with respect to fulfillment of the requirements of the Convention. **[RF]**

Answer:
The revised table of references is as follows:

Convention on Nuclear Safety **Article**	**National Report** **Chapter**
Article 6	Chapter 2
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Article 8	Chapter 3.1.3.
Article 9	Chapter 3.2.
Article 10	Chapter 4.1.
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Planned safety improvement activities	Chapter 5.4.
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Article 6 – Existing Nuclear Installations

1. It is indicated that the Gradual Reconstruction Programme (Bohunice V-1) was based on the safety report developed for the Small Reconstruction. Which organisation was in charge of this safety report ? Was it the plant operator ? **[Franc.]**

Answer:

The safety report after the Small Reconstruction was developed by Nuclear Power Plant Research Institute VÚJE. The Gradual Reconstruction was based on Preliminary Safety Analyses Report which was developed after the small reconstruction, but was focused on comparison of actual status of the plant with the requirements for the plant built according to earlier standards. The report was developed in cooperation of VUJE and Bohunice NPP.

2. Can this safety report (Bohunice V-1) be considered as a safety review using deterministic and probabilistic approaches as defined in INSAG 8 ? **[Franc.]**

Answer:

The safety report was issued in 1993 and INSAG 8 was issued in 1995 so we could not consider it as a safety review according to INSAG 8, but most of the requirements were fulfilled. Deterministic and probabilistic safety evaluation was done to set the reconstruction program requirements. The results of national and international safety evaluations of WWER V-230 plants have been taken into account.

3. More information should be given in order to establish a link between the results of the safety evaluations and the planned measures defined in the Gradual Reconstruction Programme (Bohunice V-1). In particular, does the Gradual Reconstruction Programme include the reconstruction of a safety auxiliary feedwater system? **[Franc.]**

Answer:

A lot of measures is implemented during the gradual reconstruction of the Bohunice V1 to achieve acceptable level of safety. The most important are listed in the national report. Emergency feedwater system was also upgraded. New building was built up outside the turbine hall close to the old one. Two new pumps were added to original two ones and new supply line was added to feed water trough the bottom of the SGs. Original line into feedwater line was replaced by new one leading outside the high energy lines area. **(Ref.6/21, 6/22)**

4. At present, annual licences are granted based on the progress of plant upgrading (Bohunice V-1). Is it planned, after the end of the Gradual Reconstruction Programme to grant a licence based on the results of a review by the Regulatory

Body of a complete safety report taking into account the status of the plant including all the modifications ? **[Franc.]**

Answer:
In the decision of UJDSR No. 1/1994 and No. 110/1994 there is a requirement of annual licences granted based on the progress of Bohunice V-1 plant upgrading. After completion of the Gradual Reconstruction Program, UJD SR plans to grant a licence based on the results of reviewing the complete Safety Analysis Report considering the new status of the plant in which all the modifications implemented have to be included. Anticipated service time is five years. Afterwards, Periodic Safety Review Report has to be reviewed by UJD SR again and positive results of the reviewing process will be the condition for further five years of service time. This scheme means an exception of ten year's periodicity of safety reviewing applied to other units in Slovakia. **(Ref.14/5, 17/5)**

5. It is mentioned that an Operation safety report for Bohunice Units 3 and 4 after 10 years of operation was drafted in 1993 (updated in 1996) by VUJE. Was it at the request of Slovaquia Elektrania ? VUJE is known to be the technical support of the Regulatory Body. How is the separation of roles achieved ? **[Franc.]**

Answer:
The Operational Safety Analysis Report (OSAR) was written in two steps: Rev.1 reflecting Bohunice V-2 status as of 9/1993 and Rev.2 reflecting NPP status as of 12/1996. It follows from this that Rev.2 has incorporated all design changes realized between 9/1993 and 12/1996. The OSAR after 10 years of operation was produced based on a decision of Czechoslovak Atomic Energy Commission from 1991 and completion and presentation of OSAR (plant status as of 9/1993) was a precondition for continued operation of Unit No.3 after 1995. The OSAR was contracted to VÚJE and it's subcontractors. VÚJE is the main technical support company to SE which, in some areas, is the only supplier of information in Slovak republic working for SE as well as Regulatory body. In the last two years the Regulatory body has developed it's own computational capacities, mainly using codes RELAP and MELCOR. Separation of roles is achieved in such a way that staff from VÚJE participating in creating of analysis for Slovenské elektrárne does not participate in commenting for Regulatory body. In general, reports from VÚJE presented by SE to Regulatory body are assessed and commented by the internal staff of Regulatory body. Moreover VUJE Institute was not requested by UJD SR to provide support at the review and assessment of Bohunice V-2 OSAR (final). **(Ref.8/2)**

6. Can the Operation Safety Report of Bohunice V2 after 10 years of operation (updated in 96) be considered as a complete safety review ? If it is the case, more information would be appreciated in order to establish a link between the results of this safety review and the planned measures defined in the Modernisation and Safety Upgrade. **[Franc.]**

Answer:

Yes, the Operational Safety Analysis Report of Bohunice V2 after 10 years of operation (updated in 96) is a complete safety review.

Modernization and safety upgrading program of V2 units is being realized on a „Safety concept„ basis. The sources for development of „Safety concept„ beside others are:

- recommendations resulting from OSAR V2 after 10 years of operation (updated in 1996), which are included in ÚJD SR Decision from 1996,
- safety issues defined in IAEA document EBP-WWER-03 updated for current V2 units conditions.

7. Although the defense-in-depth principle is better respected at Bohunice V2 than at Bohunice V1, PSA results are not improved (even somewhat worse). Could Slovakia explain this surprising result ? **[Franc.]**

Answer:

Original PSA results (figures) are better for Bohunice V2 than Bohunice V1. At the present time, our attention is focused on increasing the safety of Bohunice V1 units. Dominant contributors to CDF ware taken into account during preparation of the reconstruction of the plant so logically PSA results after reconstruction of the plant should be better. Very extensive reconstruction program of Bohunice V1 plant is focused on improvement of defense in depth concept to achieve internationally acceptable level of safety. Certain modification have been implemented on Bohunice V2 units, however the main reconstruction program is under preparation. **(Ref.6/9, 6/13, 6/19)**

8. Could Slovakia indicate what are the measures included in the modernisation measures between 2002 and 2010 ? **[Franc.]**

Answer:

At present, there is a „Safety Concept for V2 units„ under development, results of which will represent a basis for elaboration of design for modernization and safety upgrading.

9. It is indicated that the objective of the safety improvement at Mochovce is to reach a core melt frequency lower than 10^{-5}/year. Does this value include the risks associated with the shutdown situations and with internal and external hazards ? **[Franc.]**

Answer:

PSA studies made within the safety improvement program include also shutdown PSA and internal and external hazards. **(Ref.6/7, 6/20, 17/4)**

10. Is VUJE involved in the pre-operation safety report (POSR) as it was involved in the safety report of Bohunice ? **[Franc.]**

Answer:

POSAR is supplied by the general contractor SKODA Praha. VUJE participated also on some of its chapters. The EUCOM consortium (SIEMENS+FRAMATOM) prepared emergency analyses.

11. How does Slovakia ensure that the licensee has an adequate strategy and action plan in place to deal with the year 2000 safety issue ? **[Franc.]**

Answer:
In early summer 1998 - after approval of the „Year 2000„ project the Director General of SE, a.s. issued Direction 9/98 „Project of the year 2000„, which determine an individual managers of all the organisational units and their responsibility in this project. Projects managers were appointed, being responsible for co-ordination and informing the top management on the course of a project. Problem areas considering Y2K are identified as follows:

devices
- communication system of a company (IIS - SE)
- communication technique in operation (progr. control and safety systems)
- security systems (inc. fire protection, emergency and civil defence systems)
- ancillary systems (lifts, air-conditioning, timing switches,....)

business partners
- contractors (investments, services, spare parts supplies)
- customers (consumption, earnings, ability to pay,...)
- banks (transactions arrangement)

technology
- Hardware: servers, user´s working stations, closed control systems
- Standard software:
- operating systems
- databases
- system software
- standard software
- ancillary software
- development environment
- Application software
- user software IIS
- control systems
- safety systems
- monitoring systems
- Archived data

SE, a.s. will also conduct audit series towards its partners to ask them to declare preparedness to cope with problems of the year 2000 and their ability to carry out the contractual and partner obligations after January 1, 2000 and not later than June 30, 1999.

The project is also planned to be conducted in co-operation with Pacific Northwest National Laboratory and Argonne National Laboratory (USA). The whole process is under supervision of the regulatory body - ÚJD SR

12. Whether these curves from time to time are revised due to increase of integrated dose of fast neutrons (fluence) in the reactor pressure vessel wall and the possibilities of vessel brittle fracture? If yes, which methodology is used for the revision of these curves? **[Croatia]**

Answer:

Yes, curves are revised every year. There are detectors placed on outside surface of the reactor pressure vessel (RPV) to measure fluence of the fast neutrons on RPV. These detectors are evaluated every year during an outage and curve is corrected according to the results of the fluence. A prediction of the brittle fracture temperature is done for the next campaign.

13. Has the CDF value of 5.39 E-05 for V1 plant already been achieved? How the number could be calculated before the development of Symptom Oriented Emergency Procedures (SOEP). **[Brazil]**

Answer:

The figure shown in the National report is expected to be achieved after implementation of all proposed measures and after implementation of symptom based EOP. The figure for CDF after development and implementation of SB EOP is based on sensitivity analyses. Figures for human reliability assessment (HRA) were calculated based on Time Reliability Correlation method. Our probabilistic goal is to achieve CDF< 10 E-4. Nowadays an updated PSA study is being performed considering safety improvements and modifications realized up to now. **(Ref.6/7, 6/9, 6/20)**

14. Is it an explicit goal to meet the full set of IAEA NUSS Code and Guides, plus a set of internationally accepted industrial codes and standards (ASME, IEEE, KTA, etc.), either in fact or intentionally? Or some other defined set of regulations (e.g. US Code of Federal Regulations, USNRC Reg. Guides, German regulations) plus the appropriate industrial standards? **(Nether.)**

Answer:

In the former Czechoslovakia the "licensing basis" for construction of each design type of WWER NPPs was determined by the decision of Federal Ministry of Fuel and Energy. For example, for WWER 440-V213 reactor type such decision (issued on January 19, 1981) stipulated to use:

- 50 acts and regulations valid at that time in Czechoslovakia
- 71 Regulations and standards coming from former Soviet Union
- 77 Czechoslovak standards for design of important systems in NPPs
- 132 Czechoslovak industrial standards
- 231 additional Czechoslovak industrial standards were used, based on contract agreements with suppliers.

Safety re-assessment after some period of NPP operation is accepted principle in Slovakia, recently included also in a new "atomic law". Due to lack of codified rules and standards issued for review, assessment and upgrading process, safety requirements of regulatory authority are imposed to the upgrading programme of a specific reactor unit on the case by case basis.

"Present codes and standard" used for assessment and safety enhancement of particular reactor unit are defined in a very early stage of backfitting project. List of them is elaborated by utility (in co-operation with main contractor) and after formal application this package is reviewed and approved by the regulator. IAEA safety standards, IEC standards and codes and standards applied in Western countries are general basis for safety re-assessment and development of upgrading goals. Compliance with above mentioned list of present codes and standard "as far as reasonable achievable" is required by the regulatory body at the development of upgrading programmes. Modifications of safety important systems elaborated in this way shall be reviewed and approved by the regulatory body prior their implementation.
As far as new equipment of foreign origin installed it is permitted to adopt those codes and standards which are valid in the country where components are manufactured.

For example list of present codes and standards (for gradual safety upgrading of V-1 NPP - WWER 440-V230 elaborated by Siemens/KWU in I996) and which was accepted by the regulatory authority consisted of:
- *19 general requirements for nuclear safety (laws, regulations, rules and regulatory decisions)*
- *87 Slovak standards for Nuclear Technological Equipment*
- *6 groups of Slovak standards for electrical equipment*
- *17 IAEA NUSS standards (codes and guidelines)*
- *43 KTA standards*
- *1 RSK guidelines*
- *7 ISO standards*
- *24 IEEE standards*
- *22 IEC standards*
- *4 Interatomenergo NTD standards*
- *2 ANSI standards*
- *2 NUREG documents*
- *1 ASME code, Sect. III, Div. 2,*
- *Div. 1 of US NRC Reg. Guides*

(**Ref.: 6/16, 7/2)**

15. In this respect, do the small and gradual reconstruction programmes also address safety issues such as redundancy, reliability and physical separation (which are important items of such regulation)? Note that such aspects could imply very expensive modifications, as it may include the design and construction of additional safety trains, replacement of piping, additional diesels, etc. **(Nether.)**

Answer:
The `gradual safety upgrading" is being conducted in two phases:

a) stipulation of conceptual solutions and basic design of systems and equipment
b) detailed design and erection or modification of systems and equipment.

During the first phase (basic design) the requirements concentrate on the general project conditions such as:
- degree of redundancy
- scope of design basis conditions (design basis events)
- single failure concept
- reaction time of operators
- pipe break and leak assumptions
- safety and acceptance criteria for accident analyses.

Those conditions were - in addition to Appendix II of the Decision 1/94 of UJD and its reference document "Contract Award Safety Analysis Report for the Gradual Safety Upgrading of the V-1 Plant " issued l993 - basically derived from the IAEA Series 50 Code and Safety Guides on design. Using these guides the general requirements for nuclear safety were defined and approved by the regulatory body.

In second phase of upgrading the designers of the new or improved systems and structures may use the requirements of one of the optional foreign codes and standards, but they should also consider the original plant concept. When detailed design of individual systems and structures was completed safety important modification for each of systems was reviewed and approved by the regulatory body.

The same approach at the utilising of internationally accepted codes and standards was used at review assessment and the development of safety enhancement programmes for Bohunice V-2 NPP and for Mochovce NPP.

Some examples of extensive modifications performed or planned at Unit 1 and 2 in Bohunice:

- new dieselgenerator has been installed, replacement and realignment of cabling and reconstruction of 6 kV and 0,4 kV buses,
- full reconstruction of the HP ECCS systems were performed and new LP ECCS system was installed,
- full reconstruction of the spray systems,
- large reconstruction of emergency feedwater system,
- completely new digital RPS, ESFAS and reactor power limitation system,

16. Please clarify what licensing basis is defined, if any, and how non-compliance is treated. **(Nether.)**

(Ref.: 6/14)

17. What is your plant life extension policy in connection with Bohunice V-213 type units? **[Hung.]**

Answer:

A program is running for monitoring of the lifetime and degradation of the main plant components. Present results of this evaluation are very positive. We expect to extend the life cycle of both Bohunice V-213 units to 40 years. According to the Act No 130/1998 the Authority may extend the license for the operation of a nuclear installation on the basis of an assessment of the current state of the installation and on the basis of supplementary safety documentation.

18. Could you specify the current implementation status of Category II and III tasks from IAEA-EBP-WWER-03, „Safety Issues and Their Ranking for WWER-440 Model 213 Nuclear Power Plants„ (April 1996) for Bohunice Units 3 & 4 and Mochovce Units 1 & 2? **[Austria]**

Answer:

SE EBO prepared a material „Updated safety improvements on NPP V-2 and proposal for their solution“, which contained recommendations of the IAEA from EBP-WWER-03 applied for the conditions of NPP V-2. This material includes safety issues of I, II, III category and it is the supporting material for the Safety concept of NPP V2, which is being developed at present. The safety concept shall determine the scope of the upgrading program and safety improvement of NPP V-2.

All safety issues of Category III and some safety issues of category II were forwarded before the upgrading program itself.

<u>Status of safety issues of category III:</u>

G02 – Qualification of installations, in realization phase

CI02 – Non-destructive tests, in realization phase

S05 – Risk of blocking the filters of suction pit SAOZ, in the realization phase according to the project documentation in 1999

S13 – Vulnerability of SG feeding, in the phase of project implementation

Kont.1 – Strength behavior of the bubbler system at the maximum pressure difference at LOCA, in the realization phase

IH2 – Fire prevention, in the realization phase

IH07 – Internal risks evoked by cracking of the high-energy line, in the realization phase

EH01 – Seismic project, in the realization phase

<u>Status of safety issues of category II:</u>

SKR07 – Diagnostic systems for PO: two independent systems have been primary circuit completed, the third independent system in the realization phase of the project in 1999

E06 – Stable fire extinction equipment for the MCP deck, completed project preparation, realization to be started in 1999

E09 – Reconstruction of the air systems in the BD and ND rooms, the first phase realized – insulation of rooms, project preparation for the second phase of air system reconstruction continues.

E11 – Technical improvements for emergency regulations, phase of project preparation with the follow up realization of the safety improvement in 1999

At **SE-EMO** the measures of category III and II were primarily realized before commissioning of unit 1. In some cases when the safety measure was not finished completely, temporary measures were adopted.

19. Could you specify the current implementation status of Category II, III, and IV tasks from IAEA-TECDOC-640, „Ranking of Safety Issues for WWER-440 Model 230 Nuclear Power Plants„ (February 1992) for Bohunice Units 1 & 2? **[Austria]**

Answer:

The TecDoc 640 specifies totally 98 issues dealing with two basic problem areas - 60 issues for design area and 38 issues for operation. Based on above document the plant has introduced a large programme to upgrade the operational safety of V-230 reactors as recommended by TecDoc 640.

The progress of safety improvement is regularly monitored and evaluated by IAEA teams from 1992. According to the action plan, Unit 1 and Unit 2 are both expected to implement 92 issues related to Categories II, III and IV.

The status of the implementation of above recommendations is as follows:

Design Area :

Resolved :	Unit 1	Unit 2
category II	9	11
category III	16	23
category IV	7	10
TOTAL	32	44 issues

Under progress:	Unit 1	Unit 2
category II	11	9
category III	13	6
category IV	4	1
TOTAL	28	16 issues

Operation area:

Resolved :	Unit 1,2
category II	4
category III	6
category IV	1
TOTAL	11 issues

Under progress:	Unit 1,2
category II	7
category III	11
category IV	3
TOTAL	21 issues

20. Has the improvement in safety resulting from backfitting been quantified for the Slovakian nuclear power plants? If so, please report the results of the quantification. **[Austria]**

Answer:

Every step of the reconstruction of V-1 units was also evaluated by probabilistic approach. First PSA model was developed for pre-small reconstruction status of the plant. Later on it was modified taking into account measures implemented during the small reconstruction. The basic design for the gradual reconstruction was also evaluated by probabilistic approach. Results of this evaluation were taken into account during detail engineering of the gradual reconstruction. The PSA model is updated annually after an outage of 1st unit taking into account measures implemented during the outage. Results are shown below:

- Pre - small reconstruction status **CDF - 1.7 E-3**
- Post - small reconstruction status **CDF - 8.89 E-4**
- Evaluation of basic design **CDF - 2,85 E-4**
- Evaluation of the measures implemented till the 31.05.97 **CDF - 6.95 E-4**
- Evaluation of the measures implemented till the 31.05.98 **CDF - 4.09 E-4**

In case of safety improvement of the NPP Mochovce the deterministic approach was opted for based on Safety Issues and their Ranking, which is based on the concept of protection in depth. This approach was selected because it is more conservative. However the benefit of safety measures was evaluated also from the probability aspect. **(Ref.6/7, 6/9, 6/13)**

21. How was the function of bubbler condensers assessed in the past and what are future plans? **[Austria]**

Answer:

Within realization of safety measures great attention was paid to the issues of containment. Thorough strength and thermo-hydraulic calculations were made, complemented by experiments. Functionality of the system was fully confirmed. SE-EMO will naturally deal also with results of full-scope experiment (realized within PHARE project), however, we do not expect that it would bring significantly different results. **(Ref.14/3)**

22. In section 2.1.3 of the Report (page 21) it is indicated that the programme on safety improvement for Bogunice V-1 has been developed in such a way that to cope with the break of 2x200 mm „conventional approach„, and with the break of 2x500 mm using the best estimate method. What does the „conventional approach„ mean in the given case? What are the acceptance criteria for the core cooling used in substantiating analyses for these accidents with application of the „conventional approach„ and of the best estimate method? What are the modifications in the emergency core cooling system which provided for such considerable increase of the scale of maximum design basis accident with loss-of-coolant in comparison with the original design (32 mm)? **[RF]**

Answer:

Term „conventional approach„ is an incorrect translation. Goal of the reconstruction of Bohunice V1 is to cope with LOCA 200 mm under „conservative assumptions„ and to cope with LOCA 500 mm under „best estimate assumptions„. In this case „conservative assumption„ means that all requirements of IAEA guide EBP-WWER-01 for safety analyses (single failure, initial conditions, acceptance criteria) have to be fulfilled. To be able to achieve this goal new low pressure pumps (1+1) were installed with capacity cca 800 m^3/h each and capacity of emergency power supply was increased to 3,2 MW for each redundancy by installation of new dieselgenerator. **(Ref.6/21, 6/3)**

Article 7 – Legislative and Regulatory Framework

1. The report states that : "Environmental Offices of Regional offices issue licences concerning site selection, construction, commercial operation and decommissioning of nuclear installations based on approval issued by Nuclear Authority Ministry of Health agencies and other State administration bodies and organisations". Can the Regional Offices modify the safety conditions requested by the Regulatory Authorities for nuclear safety or radiation protection ? If no, is there a legislative disposition which prevents them from imposing such modifications ? **[Franc.]**

Answer:
(Ref.7/5)

2. In addition to the question above, it appears that the IAEA NUSS is an important reference for Slovakian reactors (as is concluded from secs. 4.5.2 and 5.3.1, p. 99). It is, however, not clear whether these, or other similar regulations, have legal status. Please explain. **(Nether.)**

Answer:
IAEA NUSS, KTA, NRC Regulations and Guides, ANSI, ASME standards and NUREG documents generally have not legal status in Slovakia. IEC and ISO standards are being step by step adopted as Slovak national standards.

As it is explained in answers to article 6 of the National Report ,internationally accepted standards or codes and standards of foreign countries can be used for safety review and for safety upgrading process based on their acceptance by regulatory body.

Licensing process is divided in the following steps:
- acceptance of list of present codes and standards for safety review
- safety concept for up-grading project and its review and approval by regulatory
- review, assessment and approval of modifications of individual systems (based on detailed design).

3. How does the general public participate in licensing processes? **[GERMANY]**

Answer:
General public participates on licensing process based on Act No.127/1994 Coll. and Act No. 50/1976 (Construction act) on environmental impact assessment. This act orders to make complete expert and public assessment of the environmental impact of constructions under preparation, including nuclear installations.

4. Which influence does it have, and is there a possibility to take legal actions against decisions of the authority? **[GERMANY]**

Answer:

Majority of resolutions of UJD SR is issued according to the law on administrative proceeding , i.e. the proceeding participant may lodge appeal against the first level resolution of UJD SR to the chairman of UJD. The exeptions are defined by .§37 of Act No.:130/1998. The next step is to apply at the court.
Similar is the situation with regard to issuing resolutions within the competency of the Ministry of health of SR for ionizing source management there is a possibility to file an appeal according to the law on administrative proceeding.

5. From section 3.1.3.1 of the Report a conclusion can be made that the license for nuclear activities is issued not by the Nuclear Regulatory Authority, but by the Regional Office. In paragraphs 4-8 of Atomic Act No 130/1998 of 01.04.98, where the procedure of licensing the nuclear activities is established, the Regional Offices are not mentioned; in the Report nothing is also said about the status, functions and subordination of the Regional Offices. Therefore it is difficult to judge if the national practice of Slovakia meets the requirements of paragraph 2 of Article 8 under the Convention. **[RF]**

Answer:

According to Act No.50/1976 (Construction act) under section 6 „Protection of special interests" it is stated that:

§126 Prior to issuing decision on siting or decision about the protection zone, building permit and completion certificate related to the construction, part of which is also nuclear installation, the Construction Authority is obliged to apply at the Nuclear Regulatory Authority for approval, which can be tied to meeting certain conditions.

This results in that without approval of UJD SR no permit can be issued for site selection, construction and operation of nuclear installation. Regional authority cannot change conditions of UJD SR, under which the approval was issued. **(Ref.7/1)**

Article 8 – Regulatory body

1. Are the salaries of the Regulatory Body staff consistent with those of operators ? Does the Regulatory Body meet difficulties in recruiting skilled staff ? **[Franc.]**

Answer:
Official comparison of the average salary at UJD and the operators, where UJD performs supervision, according to data processed by the Statistical Office SR does not give a full picture, as reporting of the average wage is given under the sector classification: „Production and distribution of energy“, and by that it does not reflect only wages of operators, but the whole energy sector. When respecting equal jobs of operators and regulators the results of comparison of average monthly salary would be higher at the operator side.

Salaries of the state health supervision employees (State Health Institutes) are lower than those in the energy sector are. Despite of that it is not very difficult to have highly qualified experts for radiation protection in the state health supervision authority. **(Ref.8/5, 8/4)**

2. What is the role of VUJE? In 3.2.2. VUJE appears as a support organization to the Licensee, and therefore subject to UJD inspections. In other parts of the report VUJE appears as a support organization for the Regulatory Body. Please, clarify. **[Brazil]**

Answer:
VUJE is a science and research organization part of which works also for UJD as a support in case of need of UJD and based on its requirements to make certain type of analyses or preparation of reviews, expertise, etc. Another part of VUJE works also for NPP operators. Based on that for some activities it has a license granted by UJD and therefore it is also supervised by the regulator. **(Ref.6/5, 8/6)**

3. How is UJD staff distributed among the several science and engineering fields? **[Brazil]**

Answer:
As of 31.12.1998 the total number of employees (including employees funded from the grant from the Swiss government) it was 79. The following table and chart give an overview of the education structure.

Education	Number of employees
Scientific degree	7
University degree	52
Complete secondary education	19
Vocational education	1

4. Could you provide some information on the financial resources of the Regulatory Body? **(Nether.)**

Answer:
The expenditures of the UJD in different years are as follows:

	1996	**1997**	**1998**	**1999**
Total (thousand SKK)	42,947	42,128	41,870	70,524
Non-investment	39,323	39,020	39,176	66,924
Capital investment	3,625	3,108	2,694	3,600
Expert services	1,932	985	1,305	2,700
Scientific contracts	9,000	6,698	5,054	7,054
Number of employees (average)	75	78	79	82

(Ref. 8/5, 8/1)

5. How are the activities of the regulatory bodies, which regulate nuclear safety and radiation protection, are financed? **[CZECH REP.]**

Answer:
The authority performs state supervision within the scope of tasks stipulated in §32 par. 1a) to e) of Act No.130/1998 Coll. on peaceful use of nuclear energy .. Radiation protection belongs under the competency of Ministry of Health of SR. Supervisory activity is performed according to §32 of the act quoted and is funded from the state budget of SR. The budget of UJD SR is compiled for a relevant budget year in a form of draft budget, which is submitted to the Ministry of Finance SR. The Ministry, based on quantification of expected revenues and expenditures of the state budget prepares starting points for compiling the draft state budget for the relevant year and negotiates with the submitting entities, i.e. also with UJD SR. These starting points are then submitted for the negotiation of the Government of SR, where the frame quantification of basic expenditure categories is determined. When the Government of SR approves the draft state budget an obligation of UJD SR is established to submit draft budget to the relevant committee of NC SR, which by its resolution gives its position to the submitted draft budget chapter. The governmental bill on the state budget is discussed by the plenary session of NC SR and is adopted in a form of an act of NC SR. Ministry of Finance distributes binding tasks, indicators and limits also for the budget chapter of UJD SR according to the adopted law on state budget. **(Ref.8/6)**

6. Does the UJD use an independent external technical support, for instance for an evaluation of the license applications or for verifying computations? **[CZECH REP.]**

Answer:

UJD has a department of safety analyses, which provides an independent technical support for UJD. This allows to make its own analyses or to check analyses, which are submitted by the operator of nuclear installation. Besides that UJD has the possibility to give assignments to external organizations or to realize analyses with the support of the IAEA or partner regulators. **(Ref.: 8/2)**

Article 9 – Responsibility of the Licence Holder

1. Has a Slovak licensee a right of recourse against his employees (in particular those in operational control) if they cause a nuclear damage either by their negligent behavior or with the intent to cause such damage ? If this the case is this right of recourse granted on the basis of the labour contracts entered into between the license holder and its employees or otherwise? **[Austria]**

Answer:
Sanctions for any damage which is caused by any employee of the operator of nuclear installations, regardless whether it was intentional or not – are imposed according to the Labor Code Nol.65/1965 Coll. (this act generally describes relations between the employer and employee in SR). Nuclear damage is subject to criminal liability and relevant provisions of the Vienna Convention. The Collective Agreement of SE (labor agreement between the employees and the company) does not deal with these issues. **(Ref. 9/3, 9/4)**

2. Is it planned to increase the amount of the license holder's liability for nuclear damages? **[Austria]**

Answer:
Since 7 June 1995 the Slovak Republic is a contracting party to:

- Vienna Convention on civil liability for nuclear damage – notification of the Ministry of Foreign Affairs SR No.70/1996 Coll.l.;
- Joint Convention to the application of Vienna and Paris conventions - notification of the Ministry of Foreign Affairs SR No.71/1996 Coll.l.;

Liabilities resulting for the Slovak Republic from these conventions are reflected in the act of NC SR No.130/1998 Coll.l. on peaceful use of nuclear energy and on changes and amendments to Act No.174/1968 Coll. on state supervision over labor safety in the wording of Act of NC SR No.256/1994 Coll.l. which entered into force on 1.6.1998. The issue of nuclear damage and its compensation is solved in §26 to §30 of Section 5 of this law. §28 par.1 gives limit of SK 2 bill. on the liability of the nuclear installation operator for nuclear damage in compliance with art. V. of the Vienna Convention.

The Slovak Republic is not considering an increase of this amount of operator´s liability in the near future.

3. Section 36 of the Atomic Act has introduced administrative sanctions against the license holder and its employees for cases of non-compliance with the provisions of this act. Has the Slovak Republic introduced a criminal responsibility of the licensee or of its employees which would have to be established by independent tribunals? If this is the case, what are the sanctions that can be imposed against the licensee or these employees? For what kind of offences? **[Austria]**

Answer:

The license holder or the employee can be placed before a court if he committed a criminal act according to the Penal Code.
Criminal responsibility, however, can be established only in relation to entity. If the license holder is a legal entity the criminal responsibility can be established only against specific employees of that legal entity – license holder, or against concrete members of the statutory body.

From the aspect of the Penal Code the action of the operator would meet the merit of the following criminal acts:
§ 97 sabotage
§ 118 unauthorized undertaking
§ 171 marring enforcement of an official resolution
§ 179 general threat
§ 181a) endangered environment
§ 187a) illegal production and holding of nuclear materials and high risk chemicals

For the above stated crimes the sanctions are mainly imprisonment (maximum a life sentence) or money penalty (from SK 5 000.- up to 5 000 000.-). If the court imposes a money penalty, it may consider the amount when imposing this penalty, if there was already a penalty imposed by the administrative authority. **(Ref. 9/1, 9/4)**

4. Does the Slovak Republic intend to become a party to the Convention on the Protection of Environment through Criminal Law (European Treaty Series/172), opened for signature in Strasbourg on 4 November 1998, which deals, in particular, with intentional and negligent offences committed by means of nuclear substances or installations? **[Austria]**

Answer:

At present the responsible Slovak authorities (Ministry of Justice and Ministry of Environment) considering the text of the Convention in accordance with the Decision of the Slovak Government No.:615/1997. **(Ref.9/1, 9/3)**

Article 10 – Priority to safety

1. In 4.1.2 the safety policy of the operating organisation, as adopted on November 17, 1997, is described. Was development of a safety policy requested by the regulatory authority, and did they assess and approve it? **(Nether.)**

Answer:

Nuclear safety has been secured primarily on the level of NPPs. Based on negotiations between the regulator and the management of the power company the board of directors of the electricity company significantly increased the attention paid to nuclear safety and prepared a safety policy which was published. The regulator appreciated this initiative, however, it was not subject to approval by the regulator.

2. Applying principles as 'In accordance with the state of the art' and 'ALARA', as declared in the Act No. 130/1998, is often difficult. Do you have (quantitative) guidelines for the application? When are e.g. modifications necessary to improve the safety level ? **(Nether.)**

Answer:

Both principles are generally declared in the Act No.130/1998, §3-(3).
Radiation protection limits are declared in the Act No. 272/1994 and its amendment the Act No. 290/1996 including personnel dose limits. Individual dose limits are based on recommendation ICRP 60. The existence of ALARA-committee at each NPP is declared in QA documentation (required and approved by UJD) as well as the rules under which every activity what can lead to individual dose higher then 1 mSv and collective dose higher then 10 mSv (5 mSv in NPP Mochovce) is evaluated, inspect and approved by the ALARA- committee. UJD evaluates safety reports (cases) and requires confirmation of ALARA principle on the basis of site specific requirements.

The operating license is usually issued for a design life time period of nuclear installations on the basis of a comprehensive safety evaluation process. For the purpose of license a justification of safety is being made with respect to current safety codes and supporting calculation. As a consequence of an engineering development throughout years it is necessary to carry out safety reassessment to comply with new safety criteria and requirements. The meaning "state of the art" declared in the Act No.130/1998, §3-(3) should be understood as a principle of keeping the safety level as reasonably achievable with respect to the latest development in the area of nuclear safety. Also requirements for periodic safety review of nuclear installations is clearly defined in the mentioned Act No.130/1998, §20-(4).

3. In the section 4.1 (page 55-58), safety policies and roles of the regulatory authority are described. What activities do licensees implement in order to infiltrate the consciousness of 'priority to safety' among persons and organizations concerned? How does the regulatory authority participate in these activities? **(Jap.)**

Answer:

There is the Nuclear and Radiation Safety Policy of the licensee Slovenské elektrárne described in the section 4.1.2 of the National Report. This policy, which is obligatory to the management and all employees, is the top document of the Company QA program for the nuclear safety area, where is expressed the overriding priority of the nuclear safety above all other Company priorities. This priority is inseminated through the Company QA programme into all nuclear safety related activities for all stages of the nuclear installation life cycle, for example:

- operation,
- personnel recruitment and training,
- procurement,
- modifications and design changes,
- radiation protection,
- technical support,
- etc.

SE, a.s. and NPPs annually state their main goals for the specific year, where the priority of safety is stressed again and safety related goals are appointed (non permissibility of OLC violation, number of events INES 1, collective dose exposure, activity of the effluents...). These goals are widely spread among the personnel through the Company and NPP periodicals, posters, periodic training.
A self checking program „SAMKO„ (analogical to STAR program) was adopted to increase the safety culture in Bohunice.

Regulatory authority requested an unambiguous attitude of the power company towards setting priorities in safety. The Board of Directors has set safety as its priority.
Regulatory authority through its inspection activity controls safety of NPP and by that also fulfillment of declared priority. Through the IAEA the regulator secured also training for NPP on safety culture for the managers.

Article 11 – Financial and Human Resources

1. Is there a formal structure (or group) assigned to feedback operational experience in the retraining programme, and to recommend modifications in the retraining programme as a consequence of design or operational procedures modifications? **[Brazil]**

Answer:

Yes, there is a special department in Slovak NPPs structure assigned to operational experience feedback.

Preparation of the theoretical and simulator training, as well as preparation of operational training is prepared for individual NPP for operational, maintenance staff and personnel of technical support annually, semi-annually, or quarterly, according to the type of training. Training programs are prepared in cooperation with the specialist for training which is at each NPP and the direct supervisor of the trained employees which respect, besides the prescribed scope of training also operational requirements resulting from realization of changes on the technology equipment, failures on domestic and foreign NPPs.

Instructors from the Training center prepare the program of training. After that the representative of NPP approves the Program. Programs of theoretical and simulator training are approved by UJD SR, which has the possibility to apply its requirements at their creation.

Upon recommendation of the Event Evaluation Committee for analysis of operational events which is the advisory body for the power plant manager, the training programs may be modified and selected employees are re-trained and trained according to them. The personnel of the power plant is trained on changes and modifications and on feedback from the operational events at whole-shift training days, which are organized once in a quarter.

Training of main control room staff during the periodical training on the simulator includes scenarios which resulted as an experience from the feedback from operational events which have occurred at the power plant, or at another power plant with reactor type WWER 440 – V 230 and V 213.

2. Do the PSA studies contribute to the planning of reactor operators training and retraining activities? How? **[Brazil]**

Answer:

Dominating sequences and identified activities of the personnel resulting from the results of PSA studies are incorporated into the training programs for the staff of unit control rooms and exercised within the basic and periodical theoretical and simulator training. Based on the evaluation, which is done every six months, the programs for the next training period are modified.

3. Are adequate financial resources available to ensure that necessary safety improvements can be performed in a timely manner (e.g., for replacement of aged equipment, modernization and safety upgrades)? **[Austria]**

Answer:
Safety upgrading programs for Bohunice NPP was mainly secured from cash flow generated by Slovenské elektrárne, a.s. There was no specific model of financing created for Bohunice Units safety upgrading programs. Company arranged several international syndicate loans for general corporate purpose in 1996-1997 and part of this resources was also used for this safety upgrading program.

Financing model for completion and safety measures implementation at Mochovce NPP Unit 1&2 is based on SKK 2,400 mil. Slovenské elektrárne, a.s. contribution and following international loans:

VÚB (Slovak Republic)	6.300 mil	SKK
VÚB (Slovak Republic)	95 mil	USD
Česká spořitelna (Czech Republic)	100 mil.	DEM
Komerční banka (Czech Republic)	200 mil.	USD
Russian Federation	80 mil.	USD
Russian Federation	70 mil.	USD
Société Générale SA	64 mil.	DEM
Kreditanstalt für Wiederaufbau	110 mil.	DEM

4. What measures are taken to assure the availability of sufficient human resources for the operation of plants? **[Austria]**

Answer:
Ensuring availability of sufficient human resources for the operation of the power plant is tied in the first place to positions with a direct impact on nuclear safety, i.e. to selected staff which have the longest and the most demanding training. To perform their function they have to pass an exam before a state examining commission appointed by UJD SR.
The number of operational personnel is set within the organizational catalogue so as to create a sufficient reserve of trained / qualified employees capable of immediate start of their function. Besides this reserve there are conditions created for preparation of other employees for selected operational functions within the annual plan of selection, training and development of human resources.

5. What measures are taken to ensure that training instructors become aware of design modifications and operational changes in a timely manner? Have procedures been developed in this respect? **[Austria]**

Answer:
All changes in the technology area are managed and organized by a regulation of quality assurance, „Changes and Modifications“. Changes in the area of documentation are governed by a regulation „Creation, approval and updating of technical documentation“.

Training of personnel on realized design changes is according to the scope of change performed in the following way (the order starts with simple changes):

a) Provable self-study of changes within the system of their introduction into the operational documentation.

b) On periodical operational training with lecturers from the technical departments (realizing the change) in more serious modifications.

c) For reconstruction works (gradual reconstruction of NPP V-1) training was realized in the following form:

- c1) theoretical training at the power plant or in the Training center
- c2) on-the-job training
- c3) training during assembly and commissioning works
- c4) training at the contractor premises.

Training center of VUJE Trnava is the supplier of the fundamental theoretical preparation and simulator training for the units in Bohunice. It has at its disposal operational documentation of NPP, including its changes and amendments. According to internal procedures instructors spend periodically stays at NPP. The organizer of the training on modifications is the Training Center together with the department of training and by the contractor's personnel.

d) Within each Contract on works relevant to design changes there is an item „training of personnel“ which specifies the content, scope and form of training.

6. Does the current legislation require the Operator to prepare and submit for the approval by the Regulatory Body a concept of decommissioning for each nuclear installation? If so, is there a requirement for a regular review of this concept? **[CZECH REP.]**

Answer:

Act 130/1998 Coll. § 14-3d, § 15-2b4, § 19-3 and § 19-4 requires the operator to prepare and submit a conceptual decommissioning plan for each nuclear installation starting from siting, construction and operation stage.

By this way a regular review is required in these important milestones of each nuclear facility life – time. In addition a regulation under preparation requires a review of the documentation for decommissioning with 10 years periodicity during operation.

7. Which body (governmental or other) is responsible for the final disposal of radioactive waste coming from the NPP's operation? **[ÈR]**

Answer:

In the SR the area of radioactive waste management, including final disposal is governed by the following laws:

- Act No.347/1990 Coll. on organization of ministries and other central bodies of the public administration of SR in the wording of later regulations:
    - based on §3 of this law the Ministry of Economy is the central body of public administration, besides other also for energy, including nuclear fuel management and disposal of radioactive waste;

  - based on §21a) UJD is a central body of public administration for the area of nuclear supervision. It performs state supervision over nuclear safety of nuclear installations, including supervision over radioactive waste management, spent fuel and other phases of fuel cycle, as well as over nuclear materials, including their inspection and registration;
- Act of NC SR No.70/1998 Coll.l. on energy and on change to act No.455/1991 Coll. on undertaking of business certificate holders in the wording of later regulations, which governs conditions for undertaking in energy sector stating that undertaking in energy sector is possible only based on a license issued by the Ministry of Economy SR. To grant such a license for undertaking in electricity and heat production in nuclear power plants it is required to have the approval of the UJD;
- Act No.130/1998 Coll.l. on peaceful use of nuclear energy and on changes and amendments to act No.174/1968 Coll.l. on state supervision over labor safety in the wording of NC SR act No.256/1994 Coll.l.. The issue of radioactive waste management is solved in §17. Based on §17, par. 12 the legal entity established or commissioned for this purpose by the Ministry of Economy SR is responsible under conditions stipulated by this law and the act of NC SR No.254/1994 Coll.l. on State Fund for liquidation of nuclear-energy installations and spent fuel and radioactive waste management. Disposal site for radioactive waste can be located only on the land owned by the state.

Based on these laws the organization responsible for safe disposal of radioactive and nuclear materials is Slovenské elektrárne , a.s. as the operator of nuclear installations.

On the January 1, 1996 the branch of SE (SE-VYZ) devoted to decommissioning of nuclear facilities, rad-waste treatment, spent fuel handling and operating of final repository was established by organizational splitting of Bohunice site. This subsidiary comply to the above requirements.

Within this branch several radwaste treatment facilities are in operation (bitumenization plant) or under construction. Major project in this field is Bohunice Radwaste Treatment Facility (BSC RAO) with following plants:

- cementation plant
- incinerating plant
- super-compactor unit

Civil construction, technology assembly, individual non-active tests of equipment and complex tests of operational systems activities are completed. Treated rad-waste will be fixed in a concrete blocks and transported to final repository placed near Mochovce NPP site. It is expected that the repository will be put in operation in 1999.

Article 12 – Human Factors

1. Additional information would be appreciated on man-machine interface, automation, organisation of plant operation, and also on operating and emergency procedures. **[Franc.]**

Answer:
To improve man-machine interface many improvements have been implemented:

Hardware modifications:
- reliability of the systems improvements
- operator actions reduction
- safety systems controls are installed on new panels
- improvement in information systems

Procedures upgrading:
- special surveillance programs developed
- new generation of normal operating procedures (user friendly, independent verification of critical steps,...)
- improvements of abnormal operating procedures
- development of Symptom Based Emergency Operating Procedures

Plant operation:
- improvement of equipment identification system
- improvement of equipment isolation and tagging procedures
- extent and criteria of post maintenance testing improvements

Since 1996 a new generation of **normal operating procedures** is under development for NPP V-2. The project is planned until 2002 and should proceed in 5 phases. Currently first phase has been completed which includes writing of 15 most important procedures in accordance with new format and structure requirements. Several modern features have adopted like step-by-step format, check lists with signatures, specific procedure for each unit and split of the procedure in description and manipulation parts. Also a new generation of Surveillance Testing Procedures was developed in V-1 and V-2 NPPs. Each test is controlled, performed and evaluated according to specific procedure and results are logged in database.

SE EBO NPP **Emergency operating procedures** are developed in accordance with methodology and in cooperation with Westinghouse Europe. The subject of the contract is delivery plant specific Emergency Response Guidelines.

The EOP are composed of two independent sets of procedures. These two groups are Optimal Recovery Guidelines and Function Restoration Guidelines. The Optimal Recovery Guidelines are entered each time, when the reactor is tripped or Emergency Core Cooling system is actuated. An immediate verification of automatic actions is performed and accident diagnosis process is initiated. When the nature of the accident is identified, the operator is transferred to the applicable optimal recovery procedure and subprocedure. These procedures are "scenario oriented" and provide recovery instructions for each accident or combination of accidents. The Function Recovery Procedures are a parallel and independent part of EOP package, which are entered when Critical Safety Function is challenged. Depending on the severity of the challenge, the transfer to Function Restoration Procedures can be immediate for a severe challenge or delayed for a minor challenge. These guidelines are independent of scenario of the accident and are based on plant parameters and symptoms. The package of guideline consists of 26 ORG (Optimal Recovery Guidelines) and of 25 FRG (Function restoration guidelines).

Validation of EOP package was performed in cooperation with Westinghouse and was in compliance with guideline INPO 83/006. Validation program was split in two phases - validation on Full Scope Simulator (FSS) and validation on Multifunction Engineering Simulator MFS). The first phase -validation on FSS was performed in VUJE Trnava in 11/97 . FSS validation program consisted of 36 scenarios.

The guidelines, which were not validated yet, because of limitations of FSS simulation, will be validated during second phase of validation - validation on MFS is taking place in the beginning of May 1999 in Dukovany.

(Ref.: 12/6, 18/4)

2. Shutdown situations have particular features concerning human factors : are there specific measures (procedures) relating to shutdown situations ? **[Franc.]**

Answer:

Each shutdown situation expected according to design is described in operational procedures. These procedures include general requirements for human actions regarding of technological manipulations in the shutdown status of the plant.

The following are the examples:

- The plant Technical specifications has specific chapter for limits and conditions in the refueling mode, which defines the requirements for system operability and surveillance, and prescribes corrective actions of plant personnel for the case of violation of operational limits and conditions.
- The Plant has a specific procedure for refueling process and manipulations with fuel assemblies.
- The Plant has a procedure for abnormal situations (event oriented) describing the recovery actions in the case of natural circulation degradation. Several situations in unit status dependence are analyzed and corrective actions are imposed in this procedures.
- The problems of foreign materials intrusion are solved in the procedure for outage of the units.
- Separate procedure exists to avoid injection of non borated water into the primary system (locking and tagging of isolation valves).
- The plant has a procedure for periodic checking of the boundary equipment to prevent the loss of coolant.

3. Several efforts have been made to reduce human failure. What hardware measures have been taken to reduce human failure and to support operator actions, e.g. does the Reactor Protection System allow a 10 to 30 minutes grace period for operator action, is a Safety Parameter Display System available? **(Nether.)**

Answer:
A 30 minutes rule has been generally applied for safety systems to cope with design bases accidents. Safety systems such as Engineered Safety Actuating System (ESFAS) have been designed to cope automatically with the short-term phase of an accident. Operators cannot make any intervention to the function of above-mentioned systems for the time period of 30 minutes unless Emergency Operating Procedure (EOP) does not require another operator's action
In order to verify a 30 minutes rule analysis and calculation has been carried out.
In order to reduce a human failure several measures is being implemented. For instance new Emergency Response Guidelines (ERG) consisting of Optimal Recovery Guidelines (ORG) and Function Restoration Guidelines (FRG), coping with whole range of design basis accidents and incorporating preventive measures for beyond design basis accident have been developed and implemented. ERG provides the operators with clear guidance to (1) identify
an event and (2) continuously monitor plant safety status.

Up to now Main Control Rooms (MCR) of operating units have been equipped with up to date plant process information system. Design requirements considering human engineering are applied for the new systems and their controls that were or will be installed as a result of upgrading process.

Design solution of the Main Control Room implemented at Mochovce Unit 1 has already incorporated features of man-machine interface. The aim of such solution was to (1) provide the operators with more convenient layout of controllers, parameters displays, different gauges, announciators, alarms, CRTs to monitor plant status effectively during various plant conditions and (2) to operate plant systems with significant reduction of potential human errors (misleading or misinterpretation of information, incorrect manipulation due to inadequate layout of controllers, etc.). Implementation of Safety Parameter Display System is under consideration in Bohunice.

For example at Bohunice V-1 units several HW modifications have been implemented with aim to reduce human failure :
- improvements of reliability of normal operation systems - controllers
- reduction of operator actions during start-up, shut-down of the unit
- implementation of reactor power limitation system
- improvements in logics of Reactor Trip System and ESFAS to minimize an operator action and to allow a 30 minutes grace period for most of DBA (for limited number of DBA the gracetime is 10 min)
- several hand valves have been replaced by electrical motor driven valves
- Post Accident Monitoring System has been installed in the control room
- Emergency Control Room has been built
- new computer process information system installed
- all safety systems are controlled from new panels in the control room
- most information which have been originally available in the non operational

(back panels) part of control room are now available in the operational part (front panels) of control room.

4. What is the policy to encourage plant staff to report „near-miss„ events on a voluntary basis? Has the experience gained through this effort been fed back into training programmes? **[Austria]**

Answer:

Official plant policy for reporting operational events results from QA documents. These describe reporting criteria for each level. The process of reporting is based on requirements of QA documents describing administrative measures for activities performance, procedures for surveillance testing, rules for equipment supervision, etc.

To reveal equipment deficiencies or adverse conditions ***management tours*** are regularly performed. Corresponding QA procedures include the division of the complete site into individual smaller tour areas (buildings, workplaces) with specified inspection frequencies for individual managers. This measure helps to cover individual parts of the plant site with tours performed by the plant managers. Detailed guidance on how to inspect and what to focus on during the inspections is included in the procedures.

To ensure that field operators report deficiencies and abnormal operational conditions a policy described in a document titled „***Field Operators Tours***“ was adopted. It includes definitions of field operators' duties when making tours of the allocated equipment and compartments. A Tour Sheet is prepared for each position.

The documentation includes written procedures of reporting deficiencies, including „near-miss“ events, detected during tours. Each Tour Sheet contains both the tour frequency and description.

There is a policy of the plant to support voluntarily reporting of inconsistencies. Such reports are analyzed by related specialized departments, resulting in numerous recommendations to inform the personnel in the frame of the regular training. The personnel is encouraged during the regular classroom training to report „near-miss“ events.

Specialized departments maintain databases of non-reportable events and they are able to provide statistic data. This process, even it is part of operational experiences feed back, is under responsibility of specialized departments.

5. What measures have been taken to assure that contractors follow plant safety culture policy (in particular contractors performing maintenance and design modification)? **[Austria]**

Answer:

According Act No. 130/1998 all contractors for NPPs in Slovakia shall have regulatory approval, which is based on complex regulatory review.

In addition contractors, to get an entry permit for NPP beside other documents, have to submit also certificate on so called general eligibility for work at NPP. Certificate is issued to employees of contractors upon proving eligibility after passing the prescribed exam at the end of a training / preparation in one of the

three groups into which the contractors are divided according to the type, location and duration of work. From all training reports are prepared and lists of trained contractors are maintained.
Training programs of contractors include also the area of safety culture. Checking whether its principles are adhered to is made by the employees of relevant departments of NPP (safety and protection of health at work, fire protection, radiation safety,..).

6. Has a systematic control room design review, including environmental aspects, been performed at the units according to international standards, and what are the results? **[Austria]**

Answer:
Safety review and upgrading of the main (MCR) and emergency control room (ECR) have been performed in compliance with requirements and criteria of international standards as IEC 964, IEC 965, IEC 1227 as well as in accordance with standards of technology supplier country (KTA 3904) following the principle "as far as reasonably acceptable".

In accordance with safety improvement of the Bohunice plant, NPP has set the goal to improve environmental conditions in the control room. There were several measures carried out to increase the safety of staff and plant equipment in case of specific events as follows:

a) in the case of fire
- installation of fireproof doors,
- installation of new breathing devices for the shift personnel,

b) in the case of earthquake
- seismic improvement of the building where the control room is situated,

c) in the case of external air contamination,
- improvement of control room air conditioning system

The quality and kind of information for operators coming from the operation processes was improved by installation of a new user friendly computer hardware and software. To improve the working conditions of the staff there were installed new tables, new floor and a digital telephone station in the control room. There is a project underway to improve the lighting, which is to be finished this year.
(Ref.: 12/1, 18/4)

7. Is the communication between plant staff and plant management at a level that assures that any concerns of plant staff are adequately and promptly addressed? **[Austria]**

Answer:
Communication between the plant staff and plant management is secured on a sufficient level through operative and working meetings at the individual managerial levels (department – section – division - plant management).

The internal directive of QA program sets process and conditions of organizing working meetings at all levels of management, OR-08. Operative meetings of divisions, sections and departments are organized minimum once a week and they deal with concrete proposals and comments of each employee.
Besides this there are so called QA boxes at the power plants where the staff can address its comments, questions and observations to the management.

Article 13 – Quality Assurance

1. The report does not mention a QA Programme before 1990. If QA programmes during design and construction of the older plants have not been applied, it can be necessary as indicated in INSAG 8 to perform supplementary testing or non-destructive examination. Could Slovakia indicate its position on this issue ? **[Franc.]**

Answer:

a. The issue of the QA program for NPP Bohunice before 1990 is explained in the answer to question 13/3.

b. On the issue of supplementary testing and non-destructive examination:

V-1 units are currently undergoing a process of gradual safety improvement. Within this process the condition of equipment was examined in detail and also supplementary testing and non-destructive examination were made within the scope which significantly exceeds the original scope of inspections.

Large volume of this type of work was made, e.g. in proving the applicability of LBB principle for the piping of the primary circuit. The scope of non-destructive examination on the primary circuit was increased; a Dutch company, RTD did independent inspection of piping of the primary circuit. Inspection of the pressure vessel of the reactor is done with the participation of SIEMENS. We bought also state-of-the-art equipment for non-destructive examination. We have introduced a program of monitoring the life of the installations and we are convinced that the condition of important equipment from the safety point of view is covered by quality inspections and testing. At NPP units we have installed stable diagnostic systems with which we are able to monitor possible leakage of the primary circuit, free particles in the primary circuit, vibrations of internal parts of the reactor, oscillation of the main circulation pumps.

Measures in the area of quality improvement and the scope of non-destructive examination, program of controlled aging and monitoring of the equipment life, installation of stable diagnostic equipment were applied also on units V-2.

We are convinced that recommendation of INSAG-8, which is mentioned by Austria, is applied at NPP Bohunice. **(Ref.13/3)**

2. What is the maximum permissible time interval between the audits for the same functional area? **[Austria]**

Answer:

According to quality manual each functional area should be assessed by internal audit at least every 3 years. In fact internal audits are done more often.

3. How were standards and regulations for quality control used before 1990? When was their use terminated? When was the integration of the new system finalized? **[Austria]**

Answer:

It is necessary to distinguish between the regulations for *quality assurance* and *quality control.*

Until 1979, quality assurance of all items of nuclear facility in Czechoslovakia based on national industrial standards valid for specific areas such as electrical, mechanical, civil parts etc. and on requirements of the Russian design.
Since 1979 the regulatory authority (Czechoslovak atomic energy commission) have been requiring licensee to establish and implement quality plans for systems, structures and components important to nuclear safety (SSCs), to assure its required quality. Detail requirements was established in regulation ÈSKAE è.5/79Zb on quality assurance of selected items in nuclear energy from the point of view of nuclear safety.
The Act No. 28/1984 on state regulation of nuclear safety of nuclear facilities, required to establish and implement Quality assurance programs and submit them for approval to the regulatory authority, to ensure quality of SSCs and thus nuclear safety of nuclear installations. Based on this act the regulatory authority issued in 1990 the regulation No. 436/1990 on quality assurance of selected items from the point of view of nuclear safety of nuclear installations requiring, in addition to, preparation, approval by regulatory authority and implementation of Quality assurance programs for each phase of nuclear installation (design, construction, commissioning, operation etc.) to assure their quality. This regulation is still in force but a new regulation is under preparation based on the Act No. 130/1998 Coll.
Since 1990 the requirements of IAEA safety standard 50-C-QA (later on 50-C/SG-Q) is taken into account by constructor and operator.

For quality control prior to 1990 we will explain in details approaches used in construction of units V-1, V-2, and after their commissioning:

A. Quality Control in construction of units V-1

NPP V-1 was constructed according to the Soviet design. Components important from the safety point of view were produced in USSR according to the Soviet standards and regulations. Manufacturing plants according to their own internal regulations performed quality control at the production. The developer (CSSR) received from the manufacturer, together with the delivery, also records on results of control at production.

The Czechoslovak party, as a developer of a nuclear power plant according to its plan, was performing entry control of equipment after being delivered to the construction site and then independent quality controls during assembly, testing and pre-operational and operational testing. Independent supervision over the quality was performed also by the bodies of state supervision.

In the inter-governmental agreement on construction of units V-1 it was stated that the construction will be realized according to regulations and standards of USSR with respecting applicable standards of CSSR. From the aspect of quality control the most important ones were the following regulations and standards of USSR:

- Regulations for construction and safe operation of nuclear power plant equipment, experimental and research nuclear reactors and sets (issued by the State committee for supervision over labor safety in industry, Moscow 1973);
- Changes and amendments to regulations for construction and safe operation of equipment of nuclear power plants (issued by the State committee for supervision over labor safety in industry, Moscow 1975);
- Standards for strength calculations for reactor parts, steam generators, vessels and piping in nuclear power plants, testing and research nuclear reactors and equipment (Moscow, Metallurgija 1973)
- Regulations for control of welded joint and weld on assemblies and constructions of nuclear power plants, experimental and research nuclear reactors and sets, PK 1514-72 (issued by the State committee for supervision over labor safety in industry at the Council of ministers USSR, Moscow 1974);
- Basic regulations for welding and weld on assemblies and constructions of nuclear power plants, OP 1513-72 (issued by the State committee for supervision over labor safety in industry at the Council of ministers USSR, Moscow 1974);

B. Quality control in construction of units V-2

For NPP V-2 Czechoslovak factories produced the decisive components. For these components quality plans were prepared (called „individual programs of quality assurance – IPZK“), and these were approved by the developer and the regulatory body. Adherence to quality plans in manufacturing equipment and during their assembly and testing on the construction site were controlled by the developer and naturally, independently also by the staff of the state regulatory body.

In the area of quality control also for units of V-2 the requirements of regulations of USSR were applied: OP1513-72, PK1514-72, standards for strength calculation (see regulations 1-5 for NPP V-1).

In those areas where there was no contradiction with the Soviet standards, the Czechoslovak standards were applied.

C. Quality control during operation of V-1 and V-2 prior to 1990

The operator of NPP Bohunice was performing operational control from the beginning of operation according to the program approved by the supervisory body, CSKAE. Operational control consists of non-destructive component testing made during the shutdown of NPP units, and from periodic testing of active systems (as for example the System of protection devices of the reactor, diesel-generators, system of emergency feed of the reactor) during unit operation with output. Records can prove all tests.

The operator of NPP Bohunice was gradually improving the program of operational controls and testing. All changes to the program were subjected to review and approval by the state supervision body. **(Ref.13/1)**

Article 14 – Assessment and Verification of Safety

1. Along with the design safety assessment seismic improvements of the entire plants were suggested. How complete is the seismic improvement of the different units? **[Austria]**

Answer:
The completion of seismic reinforcement of different units is as follows (status of 15.3.99) :

- Bohunice V-1 (Unit 1+2) 80%
- Bohunice V-2 (Unit 3+4) 20%
- Mochovce Unit 1 99.5%
- Mochovce Unit 2 75%

2. How is operating experience evaluated in order to detect precursor signs of possible tendencies adverse to safety? **[Austria]**

Answer:
The results of analysis of Operating events are regularly submitted to Event Evaluation Commttee, which is a body responsible for approval and monitoring of corrective actions.
The investigation process includes also the evaluation of recurrence of the events, i.e. monitoring of precursors affecting the safety.
The results of such evaluation are summarized in regular annual reports, in order to detect the families of precursors causing the operational events.

3. In the Mochovce nuclear power plant certain upgrades were installed in order to assure proper functioning of the bubbler condenser. Were comparable upgrades implemented also in the Bohunice V2 nuclear power plants? Are there plans for future upgrades? **[Austria]**

Answer:
Qualification of the bubbler condenser and its behavior in situations evoked by design accidents is currently assessed by the BCEQ consortium (Siemens, EDF, Empresarios Agrupados) within the PHARE project 2.13/95 „Experimental verification of the bubbler condenser“. The main task of the project is to make thermal-hydraulic testing and static tests of the steel inner-construction. NPP Paks, NPP Dukovany, NPP Rovno, NPP Kola and NPP Bohunice participate on this project. In 1997 SE-EBO in cooperation with VUEZ Levice made strength calculations for the steel inner construction of the bubbler tower. Results of calculations proved that the integrity of the steel construction will be maintained.
Starting from 1999 there is a project of internal metalic parts of bubble condenser improvement.

(Ref. 6/4)

4. Which provisions are made to ensure that plant modifications do not invalidate commissioning tests and limits? **[Austria]**

Answer:

Every safety relevant plant modification has to be analyzed and justified in accordance with the valid Slovak legislation and the plant QA documentation from the point of view of its impact on the safety margins showed in the SAR.

There are several rules established in the plant QA program. The final step of implementation of modification is the special commissioning tests. Modified systems are tested according to the modification specific procedure which includes acceptance criteria. The scope of the test and acceptance criteria for safety systems are reviewed by regulatory body. When required, an integral plant commissioning tests are performed.

The limits of the main components are maintained as defined by manufacturer. The systems specific limits and conditions are modified according to SAR and approved by regulatory body prior putting the system into operation.

The commissioning tests can not be invalidated. They can only be confirmed by new test or modified by controlled manner.

5. What is the frequency of periodic safety reviews of operating units? Are there any specific requirements for re-licensing? **[Austria]**

Answer:

According to the Law No. 130/1998 („Atomic Law„), NPP operator is obliged to perform complex and systematic safety assessment of a NPP. In a regulation, which is currently being developed, the periodicity of this kind of assessment is set as 10 years. After positive reviewing of the Periodic Safety Review Report submittal, UJD SR issues the permit for further operation of NPP unit **(Ref. 6/4, 17/5)**

6. In assessment of meeting the requirements of Article 14 under the Convention no participation of Russian organizations - designers of WWER reactors, in planning due attention is paid to the requirement for the fact that the current physical state and operation of nuclear installation continue to comply with its design (paragraph (ii) of Article 14). This is rather important having in mind the necessity to assure compatibility of the measures, listed in sections 4.5 and 5.4 of the Report, with the original design. Such compatibility could be assured only with active and implementation of the measures on safety improvement and modernization of NPP Bohunice and NPP Mochovce. **[RF]**

Answer:

The approach of Bohunice NPPs is always to operate the units within design basis. This fact was recognized also by international missions which took place since 1990.

The upgrading program was very carefully evaluated to be sure that original design requirement for operation of the plant are satisfied.

All modifications of nuclear design of the reactor has always been modified only in close cooperation with designer.

The representatives of Russian designers are permanently on site. The upgrading program is mostly focused to replace old I&C and electrical equipment by new qualified systems. The original logics of the systems are maintained in this process as much as possible. The equipment is only modified, when it results from deterministic or probabilistic analysis. The functionality of the new I&C equipment is modified to cope with newly defined design basis accident and in case when an upgrading of the safety systems is regarded. The resulting functionality is validated by quantitative safety analyses of the new design basis accident and additionally defined PIEs covering the original scope of design basis accident for the safety systems. In addition the upgrading program includes an increase of the plant ability to cope with a larger scope of LOCA by a new designed ECCS. The approach of validation of the new I&C equipment functionality covers the validation of the proper function of ECCS as well. Exactly the same approach is applied regarding the improvement of primary and secondary bleed&feed.

The modifications also consider reduction of common mode failures by improving redundancy of the systems, ensuring of separation of individual redundancies, seismic upgrading, fire protection, etc. Modified systems are thoroughly tested prior putting the unit into operation.

The negative influence of increased capacity of the safety systems (ECCS, feedwater) to the components and NSSS are evaluated by the validated computing codes which has not been available during original design. Czech and Slovak firms produced all main components, including the reactor for units in Mochovce. In order not to disturb the safety concept of the WWER 440 project the general designer – EGP in Prague, reviewed all changes. There were also consultations on solutions with the Russian institutes.

Article 15 – Radiation Protection

1. Based on the report, the national radiation protection regulation is in accordance with ICRP recommendations. Is it in accordance with ICRP 60 or ICRP 26? In the latter case, when does Slovakia intend to comply with ICRP 60 ? **[Franc.]**

Answer:
Act on protection of health of public No.272/1994 Coll.l. in the wording of later regulations is based on recommendations of the International Commission for Radiological Protection (ICRP)No.60.

2. The report indicates the average collective equivalent dose for one reactor and the evolution over the past years. The value, approximately 800 mSv in 1997, appears lower than those observed in most other countries. Could Slovakia indicate if the dose to all the workers, including maintenance operations, is taken into account ? **[Franc.]**

Answer:
The report by Bohunice NPP includes the doses from all persons who worked at the station including maintenance and operators.
The value of 800 mSv/r corresponds with the collective effective dose of WWER 440 Units in Czech Republic, Hungary and Finland.

3. Could Slovakia give more information concerning measured individual doses (average, statistical distribution, maximum) ? **[Franc.]**

Answer:
Data on questions 3 and 4 are prepared as slides.

4. The report indicates the rare gas release for EBO reactors and the evolution over the past years. Could Slovakia provide the aerosols and iodine atmospheric releases and the liquid releases ? **[Franc.]**

Answer:
Data on questions 3 and 4 are prepared as slides.

5. What measures were taken to limit or reduce exposure levels in case of accident releases? **[Austria]**

Answer:
According to the legislation valid in the SR it is the obligation of the operator and the administrative bodies on both regional and national level to prepare emergency plans. SE prepares regulations and procedures determining activity in case of accident, including procedures for evaluation and adopting protection measures during the accident. Limit values for introducing protection measures for public and control of doses of staff are fully in compliance with the recommendations of the IAEA – TECDOC-955 (August 1997). The Ministry of

Health SR prepared „Radiology criteria for evacuation and resettlement of the population in case of nuclear accident“, which were approved by the Commission of the Slovak Government for solution of radiation accidents and will be incorporated into the newly adopted generally binding regulations (laws, decrees). Emergency plans contain measures as everywhere else in the world (iodine tablets, sheltering and evacuation of persons, control of the food chain, etc.).

Article 16 – Emergency Preparedness

1. What are the criteria used to define the different measures to protect the public (sheltering, evacuation, stable iodine tablet distribution) ? What are the criteria used to enter in an emergency situation ? **[Franc.]**

Answer:
The Ministry of Health for urgent and follow up measures aimed at protection of health of population and levels of interference for ordering them based on recommendations of the IAEA applies Safety Series 109/1994. Institute of Preventive and Clinical Medicine (UPKM) in Bratislava developed a table of values for specific activities of radionuclides in food-stuffs, area activity of selected radionuclides in pasture, specific activity in fodder and values of area activity of ^{137}Cs for permanent resettlement as interventional levels.
It was based on Principles for Intervention for Protection of the Public in Radiological Emergency ICRP No.63/1993 and Safety Series No.115/1996 – International Basic Safety Standards for Protection against Radiation and for Safety of Radiation Sources (BSS). In connection with preparation of amendment to Act of NC SR No.272/1994 on protection of health of public and preparation of the Hygiene Code also this part will be amended based on recommendations of the IAEA (TECDOC 955/1997).

Criteria for implementation of individual measures according to the emergency plan are set by the staff included in the organization of emergency response based on analysis of radiation situation (identification of the source element), prediction of development of the event and permanent monitoring. Depending on the type of emergency situation the danger for the vicinity of NPP is calculated, based on which immediate measures for the surrounding public are recommended – sheltering, iodine prophylaxy. Realization of these measures is automatic after the warning and notification through radio and TV. **(Ref. 16/5)**

2. Why is area at risk in an emergency defined as 30 Km for Bohunice and as 20 km for Mochovce? **[Brazil]**

Answer:
There are 3 NPPs at the site of Jaslovské Bohunice, each having a different type of reactor. The oldest one is A-1 in decommissioning (HWGCR), another is V-1 with WWER/440-V-230 and the last one, V-2 with WWER/440-V-213. Based on data according to the analyses made by original Soviet designer the emergency area for the site of Jaslovské Bohunice with respect to NPP V-1 is set as 30-km diameter area and for the Mochovce site was set with 20km diameter.

Both areas (20 and 30 km) are set with a sufficient reserve, rounding the numbers upwards, for the whole site. At the same time also the terrain profile in the vicinity of NPP was respected.

3. Has the experience gained at the Bohunice nuclear power plant been reflected in the development of the emergency plan for nuclear power plant Mochovce? **[Austria]**

Answer:
Yes. All experience gained in introducing and exercises of the emergency plan at NPP Bohunice were taken in regard and used in preparation of the emergency plan for NPP Mochovce. Creation of emergency plan for NPP Mochovce was done in cooperation with NPP Bohunice staff, while all experience of other foreign NPPs were incorporated as well.

4. „National emergency plan is currently under development.....„ What is the current state? **[GERMANY]**

Answer:
National Emergency plan is in progress as follows:
The Commission of the Government for radiation accidents approved the proposal of UJD as for the draft and the content of the National Emergency Plan. By the end of 1999 the individual ministries will prepare meeting of NEP according to their competencies. Then the specialists from KRH SR, under the leadership of UJD, will make a review, or they will return it for re-working. The first complete draft of NEP should be ready by the end of 2000, with that KRH SR shall approve it. From 1.1.2001 it should be utilized in practice.

5. Which criteria exist for the initiation of different measures in case of an emergency? **[GERMANY]**

Answer:
Criteria based on which individual measures to protect public are applied, are derived from level of interference, which are approved by Ministry of Health. Based on that the measures to protect the public are divided into two basic groups: immediate and follow up measures.
To each type of measure a certain value of interference level is tied. These are based on regulations of the EU, ICRP, IAEA – TECDOC 955 and 953 and the safety manual of the IAEA No.109. All these criteria are respected and stated in relevant laws and related decrees valid in SR. The authors of these regulations are Ministry of Health, Ministry of Interior and UJD SR. **(Ref. 16/1)**

Article 17 – Siting

1. What are the dose limits for design basis accidents ? **[Franc.]**

Answer:
Dose limits for DBA in Slovakia are established in accordance with IAEA TECDOC 955/97 taking into account also Basis Safety Standards for Protection against Radiation and for Safety of Radiation Sources - BSS 115/96, i.e. essential dose limits are:

- dose equivalent < 50 mSv for entire body
- dose equivalent < 500 mSv for thyroid

2. Re-assessment of seismic events led to define for Bohunice a SSE of MSK 8 and to upgrade the seismic resistance of category 1 buildings and structures. What is the SSE at the Mochovce site ? Is this SSE different from the value of 0.1g and was the associated risk evaluated ? **[Franc.]**

Answer:
The SSE for EMO is 0.1 g (ZPGA) and this value is a result of a seismic risk evaluation for EMO-NPP area. A tectonic stability and design basis ground motion parameters for the Mochovce site were reviewed by IAEA Seismic Safety Review Mission in 1998 year upon the request of Nuclear Regulatory Authority of the Slovak Republic. **(Ref. 17/3, 17/4)**

3. At the Mochovce NPP, the horizontal acceleration for earthquakes has been increased from 0.06g to 0.1g.
Is the new value for the horizontal acceleration appropriate with respect to the seismic situation? What measures are planned to protect equipment and buildings important to safety? **[GERMANY]**

Answer:
Yes, the new value appropriately respect the seismic situation. Additional reevaluation of seismic resistance of all safety important buildings (cca 15) and equipment (cca 17 500) was performed. Identified equipment, which did not meet acceptance criteria, were reinforced or replaced by qualified ones. Only minor local reinforcements of buildings was needed. **(Ref. 17/2, 17/4)**

4. Have state-of-the-art seismic analyses been performed for the Mochovce and Bohunice nuclear power plant sites (e.g., probabilistic seismic hazard analyses, seismic margin analyses, seismic PSA)? Did any upgrading result from these analyses? If the analyses are planned, what is the schedule for their completion? **[Austria]**

Answer:
Following methods were used for the seismic re-evaluation program at Bohunice site:

Seismic qualification by Earthquake Experience

The seismic experience based methodology developed by the Seismic Qualification Utility Group (SQUG) for verification of seismic adequacy of existing NPPs. This methodology is intended to complete the design basis of plants whose equipment was not seismically qualified. This methodology is applicable to generic classes of equipment. It does not cover structures or piping. A generic implementation procedure (GIP) provides detailed criteria for implementation.

A probabilistic safety analyses
In the PSA method the postulated seismic event over a range of probabilities which typically range between 10^{-3} to 10^{-5} per year is defined as an initiating event. The SSC, which are identified by deterministic and probabilistic approach as potentially significant for safe shutdown are included in the Safe Shutdown Equipment List (SSEL). The SSEL are finished for Bohunice V-1 and V-2.

A seismic margin assessment (SMA)
The seismic margin assessment method has typically been used for seismic re-evaluation of existing facilities for beyond design basis earthquakes in the international community and is presented by EPRI in the NP-6041 document.
This method we partially used at V-2 units.

A new comprehensive state-of-the-art probabilistic seismic hazard analysis has been performed for Bohunice site in 1996-98. The analysis has been performed in accordance with IAEA Safety Guides (1991, 1994), IAEA Technical Guidelines (1996), US NRC RG 1.165 (1997) and other relevant documents. The analysis has been performed in the following steps:

- analysis of previous estimates of the seismic hazard for Bohunice site,
- compilation of the seismological and geological databases,
- determination of attenuation,
- probabilistic computation of the uniform hazard spectrum,
- determination of Review Level Earthquake (RLE) characteristic.

The analysis was performed under supervision of the K.W. Campbell and R.D. Campbell of EQE International. It was also a subject of the IAEA Seismic Safety Review Mission to Slovakia in November 1998.
The analysis led to the proposal of the new RLE characteristic (the RLE corresponds to the SL-2 in IAEA, 1991). The RLE for the Bohunice site is characterized by

- magnitude and distance of the controlling earthquake for the 0,2 s UHS value (the UHS spectrum corresponds to the probabilistic mean 10 000-year response spectrum),
- horizontal and vertical response spectra (including the PGA value) for 5% of critical damping,
- sets of three component accelerograms, whose spectra were matched to the horizontal and vertical RLE spectra.

Following upgrades resulted from the calculations and review of the seismic demand for structures, systems and components at Bohunice V-1 units to reduce seismic vulnerabilities of equipment to a minimum:

- reinforcement of existing connections
- implement additional supports(i.e. for piping)

- replacement of existing supports and/or anchorages
- application of corrective measures to existing support structures
- reconstruction of the main building complex, emergency pump house, dieselgenerator building, stack, brides and administration building.

For Mochovce NPP the seismic input data and seismic qualification of SSC were reevaluated.

Seismic input data
Based on the new requirements from NRA (ÚJD-SR) to follow IAEA guide 50-SG-S1, rev. 1, was the input data for the EMO NPP reevaluated. Previous works, were done:

- Study - Supplementary review of seismic hazard and faulting in area of interest of EMO NPP - by Dr. Simunek & CO, EGP Praha, 1992,
- Evaluation of works in the area of interest of EMO NPP - by Dr. Juhasova & CO, SAV Bratislava, 1994
- Geological evaluation of EMO NPP site, Environmental and geological survey - EQUIS, 1996
- Seismological database for EMO NPP, final report (reviewed edition),- by Dr. Moczo, SAV Bratislava 1997

The results were completed in the POSAR, chap. 2.5, made by EGP Praha, as general designer for EMO and the main responsible organization (contractor) for siting and Basic design, and were finalized in 09/97.
The evaluation of seismic hazard was done by deterministic methodology. In frame of this work was made estimation of recurrence period of maximal potential earthquake. The comprehensive work concludes from following parts:

- completion of seismological and geological database
- reassessment of seismogeneric domains
- attenuation determination
- establishing of hazard spectrum and iRLE
- calculation of recurrence period for OBE and SSE level earthquake

The POSAR was submitted to ÚJD SR 12/97.
This works indicated increasing of SSE (SL-2) level (ZPGA) from 0.06 g to 0.1 g in horizontal direction (and 2/3 in vertical d.) with probability of such events 10^{-4}/year. To this value was associated standard response spectra NUREG-0098 with 5% dumping. From the input data three directional accelerograms and next the new floor response spectra were calculated.
The result are demonstrated on IAEA mission in 11/98. The major results were, that **used methods and the associated results are accepted and are in accordance with international practice**. Also was recommended minor investigation for next future and alternative probabilistic calculation for better demonstration of safety margin of nuclear installation.

Seismic qualification of building and equipment

According to the increasing of SSE level, in frame of „safety upgrading project“, was reevaluated the seismic resistance of all buildings and equipment which are important from nuclear safety point of view.
As base for reevaluation method was used the SQUG method developed for the existing nuclear installations mainly in US and West-Europe.
Also was included a generic implementation procedure (GIP), and on the request of SE, a.s. was made in frame of technical cooperation program with IAEA basic document - „Technical guideline for seismic reevaluation of EMO NPP“.
In the frame of classification was created a group of nuclear safety important equipment (so called nuclear island), and was screened the ability of withstanding against the SSE level earthquake - for shut-down, cooling-down and state in that position min 72 hours (residual heat removal).
All seismic important building were recalculated. The evaluation indicated only minor, local reinforcement in reactor building and auxiliary building and feasible reinforcement in service building. All reinforcement was completed to start up of 1 unit.
The technological part was reevaluated, and at this time (03/99) was reinforcement or replacing done to 98% for unit 1 and 70% for unit 2. All this work should be finished at the end of 1. refueling outage in unit 1, resp. start-up for unit 2. **(Ref. 6/9, 6/20, 17/2, 17/3,)**

5. What are the requirements for nuclear installation operators to periodically re-assess external hazards and the adequacy of facility design against those hazards? Does this extend to both man-made external hazards as well as natural phenomena hazards? Which verification processes and monitoring programmes are in place for ensuring that relevant new external hazards information is brought to the attention of the regulatory body and the nuclear power plant operators on a timely basis? **[Austria]**

Answer:

Assessment of external hazards man-made and caused by natural phenomena are currently included in the OSAR NPP V-2 which follows the content and format established in US NRC guide REG.1.70. Reassessment of external hazards and adequacy of the design will be performed periodically every 10 years within the Periodic Safety Review, according to the decision of the Regulatory body from 1996. Following hazards have been analyzed in NPP V-2 OSAR :

Man induced external events:

- airplane crashes
- damage to major pipelines in the vicinity of the NPP site
- missiles in turbine hall jeopardizing containment

Natural phenomena:

- external flooding
- wind loads
- seismic events

New OSAR of V-1 after reconstruction is currently under development and all of the above given hazards will be included.

Evaluation of potential to man induced external events caused by new installations on-site is insured by the QA principle requiring safety assessment of all design changes, among others, also in the Department of Nuclear Safety which is responsible of maintaining and updating of OSAR. If relevant, appendix to OSAR has to be prepared within the modification process.

Periodical reassessment of the external hazards at Mochovce NPP
Based on the scope of „Safety problems of VVER type reactors“ (doc. IAEA SC-108) was prepared and implemented the „Safety updating project“ in EMO NPP. In frame of this project was reviewed next topics:
Man induced external evens:

- aircraft crash
- shock wave from potential explosion (hydrogen, natural gas)
- missiles
- heavy load drops

and natural phenomena

- external flooding
- extreme wind loads
- extreme temperatures

All this investigations, analyses was made and their conclusion was implemented in EMO NPP nuclear installation.
As base of the analyses was adopted IAEA guide 50-SG-S11A with recurrence period 10^{-2} and 10^{-4}/ year.
There was elaborated by EGP Praha (General designer and responsible organization for siting of NPP) the „Study of meteorological condition in area of interest for EMO NPP“.

As results of this analyses some additional modifications were implemented and the results were included in the new version of the POSAR (from 1998).
The assessment will be repeated on the bases of periodical review of safety and will be reflected in relevant safety documentation (mainly in OSAR) in the recurrence period 10 year. Also, if there will be special request of ÚJD SR it can be repeated at between this term.
(Ref.6/4, 6/20, 14/5, 17/2, 17/3)

6. Which national standards does the regulatory body apply and to which extent do these standards comply with international standards with regard to the design of nuclear installations against external hazards? **[Austria]**

Answer:
Principles (conditions) for application of regulations and standards in design of the former Czechoslovak NPP with WWER-440 units with V-213 reactor type are published in the Measure No.3/1981, Ministry of Fuels and Energy of CSSR, issued based on a resolution of the government of CSSR No.303/1979, item III/1. These principles include an extensive list of norms, standards, directives, decrees

(both Soviet and Czechoslovak) used in preparation of both technical and implementation projects for NPPs.

The specific list of international standards and criteria used in preparation of pre-operational safety report of NPP EMO and proving nuclear safety of NPP EMO against external impacts is stated in individual sub-chapters of the pre-operational safety report of NPP EMO (sub-chapter 3.2 to 3.13):

1) Technical Guidelines for the Re-evaluation Program of Mochovce NPP (units 1-4). IAEA, Vienna, August 1995.
2) IAEA Safety Series 50-SG-S1: Earthquake and Associated Topics in Relation to Nuclear Power Siting. IAEA, Vienna, 1991
3) IAEA Safety Series 50-SG-D15: Seismic Design and Qualification for Nuclear Power Plants. IAEA, Vienna, 1992
4) Criteria for Seismic Evaluation and Potential Design Fixes for VVER Type Nuclear Power Plants. Prepared for IAEA by Stewenson and Associates, Cleveland, 1994
5) Masopust R., Podruzek J.: Requirements for re-evaluation of seismic resistance of constructions and installations of NPP EMO, unit 1 and 2. Technical report SKODA PRAHA and Stewenson and Associates, arch, c.Jc 42075Zp/Rev.4 or REP 15-95.SPH/Rev.4, Plzen, 1997
6) Resolution of CSKAE No.2/1978 on securing nuclear safety in design, permitting and realization of constructions with nuclear energy installations, CSKAE, Praha 1978
7) Decree CSKAE 436/1990 on quality assurance of selected installations from the point of nuclear safety of nuclear installations
8) IAEA Safety Guide No.50-SG-S11A, Extreme Meteorological Events in Nuclear Power Plants Siting Excluding Tropical Cyclones, Vienna 1981
9) STN 73 0035: Load on civil constructions, M.Tichy a kol.
10) Directives for calculation of break wave pnB 121/1971 VUVH Bratislava, 1972
11) ANSI/ANS 2.12: Guidelines for Combining Natural and External Man Made Hazards at Power Reactor Sites. ANSI/ANS, 1978
12) IAEA Safety Series 50-SG-S3: Atmospheric Dispersion in Nuclear Power Plant Siting. Part B; Extreme Meteorological Conditions in Nuclear Power Plant Siting. Vienna 1980
13) STN 35 0001 – Shielding of electric machines rotating
14) IAEA Safety Guide No.50-SG-D4: Protection against Internally Generated Missiles and their Secondary Effects in Nuclear Power Plants, Vienna 1980
15) IAEA Safety Guide No.50-SG-D1: Safety Function and Component Classification for BWR, PWR and PTR, Vienna, 1979
16) IAEA Code of Practice No.50-C-D: Design for Safety on Nuclear Power Plants, Vienna 1978
17) Regulatory Guide 1.115, Protection against Low-Trajectory Turbine Missiles, US NRC, Regulatory Guide, Revision 1, July 1977
18) Design of Structures for Missile Impact, Topical Report, BC-TOP-9A, Rev.2, Bechtel Power Corporation, California, September 1974.

Article 18 – Design and Construction

1. With reference to defense-in-depth, prevention and mitigation of accidents, what were the design problems, and what were the main reasons underlying the modifications implemented or planned ? **[Franc.]**

Answer:

The plant upgrading program has been based on both national and international safety evaluation programs.

The improvements in the prevention of accidents are mostly focused on :

- Emergency operating and maintenance procedures and rules
- ISI and surveillance program
- fire prevention

The improvements in prevention of accidents were :

- extension of DBA (LB LOCA, MB LOCA, MSLB, Earthquake, Fire...)
- setting of preventive modification AM measures for BDBA - in the EOPs including HW modifications
    - electricity power supply (devoted)
    - feed and bleed on primary and secondary sides
- improvement of redundancy or separation of safety systems
- single failure criterion application
- common mode failures elimination as practicable achievable

(Ref.: 18/2)

2. It is stated on p. 74 that NPPs are built according to the defence-in-depth principle. Does the CSKAE Decree No. 2/1978 On Securing of Nuclear Safety in Designing, Approving and Implementing of Constructions with Nuclear Power Installations also require compliance with such high-level safety philosophies or is the requirement to fulfil the defence-in-depth principle only indirectly given by demanding that all technical requirements have to be met (see sec. 5.2.1 of the report)? **(Nether.)**

Answer:

The decree establishes requirements for „ensurance of nuclear safety of nuclear power installations during design, approval and construction with the aim of implementing of uniform principles of state technical policy in the construction and the environmental solicitude."

The technical requirements included to Regulation No. 2/1978 issued in 1978 are in compliance with earlier version of US 10 CFR Part 50 App. A. The principles of defense in- depth were published by IAEA later when the INSAG-3 report was elaborated. The principles of defense in-depth are implicitly covered by the technical requirements included in the above mentioned regulation. **(Ref.: 18/1)**

3. As mentioned above, it is stated that nuclear reactors in Slovakia are built according to the principle of defence-in-depth (p. 74). A corollary is that they must

satisfy the multi-barrier principle (see e.g. IAEA Code on Design). Although important improvements have been made on the leaktightness of the confinement of the VVER 230-type (p. 21), a leakage rate of still more than 50% per 24 hrs. conflicts with the multi-barrier principle and, hence, with the defence-in-depth principle. In this sense, there appears to be no full compliance with Art. 18.
What method has been used to calculate the source term (the release to the environment) from design basis events (e.g. USNRC Reg. Guide 1.3 or 1.4, German methods)? **(Nether.)**

Answer:
Review and assessment of design of WWER 440/V-230 reactor type accomplished by experts nominated by IAEA in the frame of Extrabudgetary Project were based on defence-in-depth safety philosophy. Defence-in-depth approach was directly included in the clasification criteria (see IAEATECDOC 640).
Safety upgrading program for Bohunice V-1 Units reflects all of TECDOC 640 recommendations and in many aspects goes beyond usual way of addressing of safety issues and even beyond recommendations of TECDOC 640.

The limiting cases for source term are the LOCAs. For these events by accident analyses codes (RELAP-5, TRAC or CATHARE) the fuel cladding temperatures and pressures in primary circuit are calculated which serve as an input values for fuel damage prediction (codes FEMBUL-2, DEFOS-A). Based on the predicted failed fuel amount the release term is determined to the coolant. Due to the fact that fuel temperatures during this accident are less or equal ones during normal operation the released activity to the coolant is taken as the gap + plenum inventory in the failed fuel on the end of the fuel cycle.
The release ratios to the environment are calculated as ratio of fission product + corrosion products contained in the released air or steam to the mass of these substances (air, steam) contained in containment.
Radioactive iodine release is assumed in three forms - aerosols (90 %), organic (7 %) and elementary form (3 %).

The source term for Beyond Design Basis Accident (guillotine rupture of the cold leg of the main circulation line, 2 x Φ500 mm) has also been calculated in accordance with the methodology described above and is described in the Preliminary Safety Analyses Report.

4. How is an easily manageable operation with regard to the human factor aspects guaranteed? The last sentence of sec. 5.2.1 reads: "The human factor is only considered with respect to activities outside of the nuclear installation". What does this mean? Please explain how ergonomic principles, man-machine aspects, etc., are recognised in the control room design. **(Nether.)**

Answer:

There is a mistake in the sentence "The human factor is only considered...". Right wording should be "The human factor is also considered..." and safety re-assessment and development of upgrading program for MCR were meant.

Human factors had been considered at the initial stages of the reconstruction project and throughout the design process to assure that the functions allocated to the operator[s) and maintainer(s) can be successfully accomplished to meet the safety system design goals. Adequate administrative controls and security are being provided to prevent unauthorised changes from being introduced through a human-machine interface (e.g. authorisation to open cabinets, use of keylock controls, restriction on vital area access, etc.). Administrative controls and design features have been specifically addressed as for software access in addition to typical equipment access provisions. The aim of all of these provisions is to minimise the possibility of system (especially safety systems) failure due to human error, or due to unauthorised entries or alterations of the system through a maintenance, test, or configuration interfaces. These types of failures are also considered in the failure analysis.

Safety review and upgrading of the main (MCR) and emergency control room (ECR) have been performed in compliance with requirements and criteria of international standards as IEC 964, IEC 965, IEC 1227 as well as in accordance with standards of technology supplier country (KTA 3904) following the principle "in compliance as far as reasonably achievable".

Design features, requirements and criteria of these standards contain both human-machine interface and ergonomic principles and aspects.

(Ref.: 12/6, 12/1)

5. Which additional safety analyses have been performed to reflect the ongoing process of design changes and facility upgrading measures? **[Austria]**

Answer:

Bohunice V-1 units:

Safety upgrading of V-1 units has been supported by a large number of safety analyses performed in all stages of the project life: Basic engineering, preparation and licensing of „small reconstruction„ as well as „gradual reconstruction„ programs. To support basic design of the gradual reconstruction, LOCA and non-LOCA analyses were done by Siemens. To verify correctness of detail design of safety systems, new safety analyses were developed in 1998 by independent company Energoprojekt Sofia. Many other partial safety analyses have been done during preparation of the reconstruction.

All decisions on design changes have been based on evaluation of contribution to safety in deterministic and probabilistic areas. Any change to original design presented to Regulatory body within the licensing procedure had to be supported by safety evaluation and modification of specific part of the OSAR and Limits and Conditions included. Reconstruction have resulted in substantial revising of OSAR (status after small reconstruction) and currently a completely new OSAR is being produced by consortium Siemens-VÚJE, in compliance with US NRC Reg. guide 1.70 and IAEA guide EBP-WWER-01. Additional analyses dealing with severe accidents, accidents in shut-down modes (EBP-WWER-09) and PTS analyses (EBP-WWER-08), which are not required in the US NRC document will be attached.

Examples of areas extensively analyzed:
- completely new design of ECCS
- new logic and signals in the reactor protection system
- improved design of Accident localization system

Bohunice V-2 units:
Project of safety upgrading of V-2 units is currently under development. All design modifications in project will be supplemented by new safety analyses, new database and specific safety analyses will be performed in support most important design changes.

6. Have the codes used for these new analyses also been modified? **[Austria]**

Answer:
The thermohydraulic codes used for safety analyses are being updated continuously by the contractor organisations within their own development and QA program. In general, all organisations working for EBO use latest or recent versions of codes. VUJE as main engineering support company even participates in development programs of codes, like CAMP for RELAP5 and takes part in all engineering projects supported by IAEA. Based on the decision of the Regulatory body from 1996 the information, validation and verification of codes used for analyses in OSAR has to be included into OSAR. Quality of codes available is an important factor in decision process in the frame of contracting of safety analyses.

In the frame of safety upgrading of V-1 NPP new models have been developed for RELAP5 and MAAP4/VVER codes. In general, the quality of software used for safety analyses of NPPs V-1 and V-2 is very good and quite comparable with the situation in western Europe and USA.

7. How is the need for improvement decided upon? **[Austria]**

Answer:
The safety of the **Bohunice V-1 plant** has been evaluated extensively by various organizations, including (Czecho-)Slovak and international bodies and missions, against western and international safety standards. These evaluations resulted in a number of requirements and recommendations for the enhancement of the safety of the plant.

The former CSKAE (Czechoslovak Atomic Energy Commission) in its decision No. 5/1991 extracted from these requirements and recommendations those improvements in safety that had to be implemented in the two stages of the reconstruction of the plant. The first stage, the so-called small reconstruction brought the improvements (81 + 14 short term backfitting measures) necessary to continue of plant operation. The small reconstruction was completed in 1994.

The second stage (so-called gradual reconstruction) , required by UJD SR´s decision No. 1/1994 planned for the period of 1996-2000, has brought further essential safety improvements allowing the continuation of plant operation.

At present, a project of modernisation and safety enhancement of the **Bohunice V-2 plant** is being developed. This is based on results of safety re-assessment after ten years of operation, as well on extensive evaluation of the plant safety carried out by various national and international companies and missions. Decision on safety enhancement has been made and codified in UJD SR decision No. 4/1996.

Regarding the **Mochovce NPP** the operator accepted all the recommendations and measures for the plant safety enhancement, specified by a number of international missions, mainly in IAEA and RISKAUDIT recommendations. Most of the recommendations and safety measures have been implemented so far. The rest of them is requested for implementation in ÚJD SR decision No. 318/1998 as a condition to the operating licence.

Some sources for safety upgrading programs are listed in Annex 6.3 of the National Report.

Article 19 – Operation

1. Is there in the Operation limits and conditions (OLC) document which is part of the Safety report, the Reactor Coolant System Heatup and Cooldown Limitations rates? If yes, which are the highest allowable rates (°C/hour)? **[Croatia]**

Answer:
Operational limits and conditions are a part of the Operational Safety Analysis Report. The limitations of the primary coolant temperature changes for normal operation are following:

- temperature increase less than 20 °C/hour
- temperature decrease less than 30 °C/hour

2. This section enumerates the different tasks (14) of the technical support and safety units that are parts of the licensee's organisational divisions. Task n° XIII is related to the organisation and co-ordination of liaisons of the divisions with the Regulatory Body in the field of nuclear and technical safety. Which of the above tasks relevant to safety require the approval of the Regulatory Body when the plant is in commercial operation (after issuing the licence) according to the Act of the National Council of the Slovak Republic No 130/1998 ? **[Franc.]**

Answer:
Following tasks relevant to safety are subjects of regulatory body approval in accordance with the Act No.130/1998 when the plant is in commercial operation:

- Item I - Regulatory requirements on radiation safety are not explicitly covered by Act 130/1998 since this competence belongs to Ministry of the health. However, radiation safety is also covered by §20 Nuclear Safety of the act 130I1998.

- Item IV - Procedures for normal and emergency situations (OP and EOP) are submitted to UJD for review. This is covered in §15 section 2 of the Act 130/1998. UJD can also impose the licensee to make modifications or improvements in content and format of OPs and EOPs.

- Item V - Regulatory requirements on supervision over the nuclear safety in nuclear installations is covered by §32 of the Act 130/1998. Modifications to be carried out at nuclear installation are subject of regulatory body approval according to § 20 section 5 of the Act 130/1998.

- Item VI - Regulatory requirements on Event analysis and experience feedback are covered by §24 of the Act 130I1998.

- Item VIII - Regulatory requirements on Surveillance testing of safety and safety related systems are covered by §20 section 9 of the Act 130I1998.

- Item IX - Regulatory requirements on safeguards are covered by §12 of the Act 130/1998. Regulatory requirements on fuel cycle, calculation of the core and regulatory supervision are covered by §15 of the Act 130/1998.

- Item XIV - Regulatory requirements on Emergency planning e.g. management and organisation are covered by § 25 of the Act 130I1998.

3. Information would be appreciated on the safety classification of radioactive wastes for disposal and on the corresponding criteria. **[Franc.]**

Answer:
There is no classification system of radioactive wastes in regulatory documentation so far and regulatory requirements in the Slovak Republic follow the classification system proposed by the IAEA.

The classification system for radioactive wastes is proposed in the new ÚJD SR regulation prepared as a respective regulation to the new atomic act. According to this classification radioactive wastes are divided into 3 categories - very low level, low and intermediate level and high level wastes. The category of low and intermediate level wastes is further divided into 3 subcategories.

The low and intermediate wastes will be disposed of in a near surface disposal facility located near Mochovce NPP. The acceptance criteria were derived on the basis of safety analyses assessing the normal and intruder scenarios of radionuclides transport to the individual of critical group. The proposed criteria includes the activity concentration for 19 radionuclides.

4. Do sub-contractors who operate waste disposal systems have a specific license? **[Franc.]**

Answer:
The near surface disposal facility for radwaste according to Act No. 130/1998 is an nuclear installation and for its siting, construction and operation specific licence is required. The radwaste disposal facility will be operated by Slovenské elektrárne, a.s. company - branch SE-VYZ.

5. How are the Symptom Oriented Emergency Procedures developed in cooperation with Westinghouse being validated? **[Brazil]**

Answer:
In the first stage of contract preparation the applicability of Westinghouse generic analyses for V-213 NPP has been assessed based on a set of representative analyses. The result was that most of the strategies in Westinghouse EOPs can be applied in V-213 EOPs. Individual strategy changes have been supported by specific analyses performed concurrently with the development of EOPs.

After finalizing the EOP a two phase validation process is in place. Validation of EOP package was performed in cooperation with Westinghouse and was in compliance with guideline INPO 83/006. Validation program was split in two phases:

a) validation in Full Scope Simulator (FSS) in Trnava (36 scenario) with plant staff from Bohunice and Dukovany was successfully completed in 11/97, findings have been already resolved and EOPs revised
b) validation of strategies that could not be validated on the full scope simulator (because of extreme parameters and FSS limitation) will be validated using Multifunctional Simulator (MFS) developed in the frame of PHARE project. This part of validation would be performed in 1999 in Dukovany.

At SE-EMO the EOPs were validated on a full scope simulator at NPP Mochovce with the participation of WESE experts. Validation was made with the participation of two MCR crews in October 1998. **(Ref.: 19/9)**

6. The US NRC requirements for Operation Limits and Condition (OLC) are based on the existence of Standard Technical Specifications for the various suppliers. How were this Technical Specifications adapted to WWER-type reactors? Are the OLC for V-2 plant available for the public? **[Brazil]**

Answer:

Operation Limits and Conditions for Bohunice units were developed in 3 phases:

Phase 1 - based on different information sources (design documentation, procedures, Preliminary Analysis, general safety requirements of Russian origin). OLC for NPP V-1 were developed in the frame of plant licensing.

Phase 2 - new format and structure of OLC were adopted, using US NRC Standard Technical Specifications for Westinghouse NPPs as a format and content template. Within NPP V-2 licensing OLC requirements were collected from design documentation, Preliminary Safety Analysis Report of V-2, OLC for V-1, general safety requirements and procedures. Czechoslovak General Constructor (SKODA Plzen) was responsible for this development.

Phase 3 - OLC for NPP V-1 were updated to structure and format of V-2 OLC

Upgrading of OLC:

Development of bases for OLC was a state funded R&D project coordinated by VUJE Trnava and some of the technical specifications were qualitatively justified. Currently a new project for OLC improvement (wording, logic, comprehensiveness, unambigousness, ..) and justification (development of deterministic and probabilistic bases) is underway for NPP V-2.

OLC for the Unit 1 in Mochovce were prepared by the general supplier ŠKODA Praha. They are part of the SAR and they are based on the design of units VVER 440.

OLC are available for the public. **(Ref. 19/10, 19/7, 19/12,)**

7. It seems that the OLC uses the definition of „Modes„ of operation (as in the US NRC Reg guide 1.70). How many modes of operation are defined for V-2 plants, considering the possibilities of operation with one or 2 isolated loops? **[Brazil]**

Answer:

In the OLC seven modes of operation are defined but there is no link to the status of loops. The OLC in principle does not prohibit operation with one or two non.operated loops. For operation in the mode 1 (power > 2% of nominal power) the maximum allowed power level is limited by number of operating loops.

Major difference in definition of operating modes used in Slovakia from mentioned US NRC RG is the mode No 7 additionally used in Slovakia. This covers plant conditions with reactor without fuel. It reflects RPV ISI Program with 4 years periodicity when all fuel is discharged from reactor.

(Ref. 19/10, 19/6, 19/12)

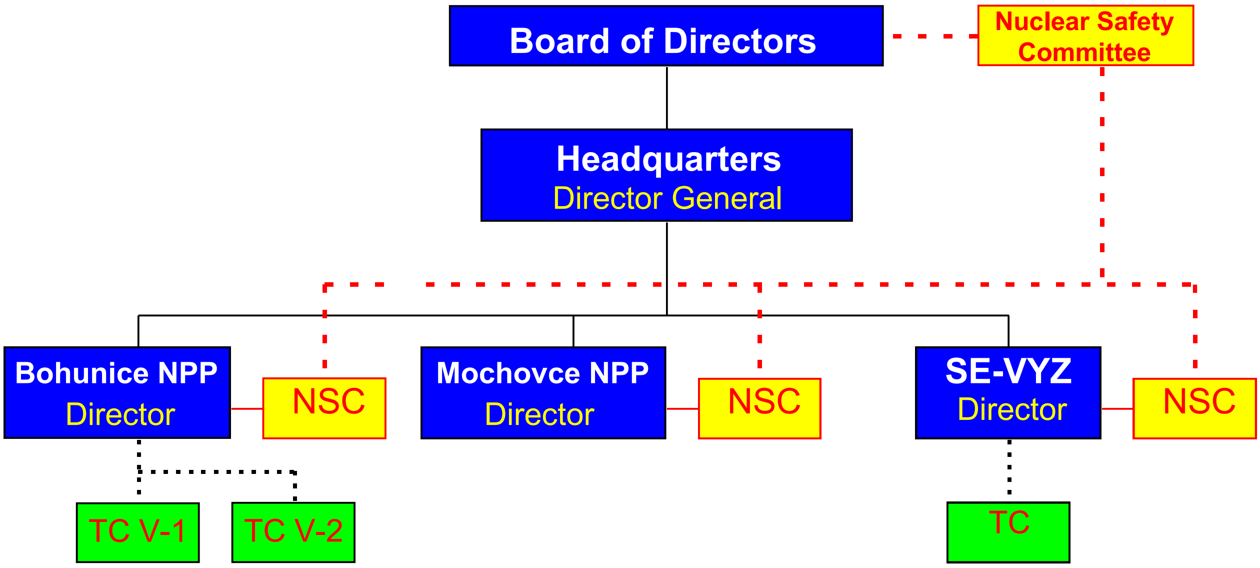
8. Please clarify the role of the „Nuclear Safety Committee„ mentioned in the text. Is there one or more committees? What is the composition? **[Brazil]**

Answer:

The Nuclear Safety Committee of SE is an advisory body for the Board of Directors of SE, which assesses and proposes solutions for generic safety issues of nuclear installations of SE. The Committee is composed of members from the top management of the joint stock company Slovenské elektrárne, and its subsidiaries operating nuclear power installations, further representatives WANO, universities and research institutes dealing with use of nuclear energy and experts in Slovak Republic.

The idea of establishment of this committee originates from 1995 when the Nuclear Safety Committee was established at NPP Bohunice as an advisory body for the power plant manager. This committee started to work in 1996. Similar committees are constituted and are working also at other subsidiaries of SE operating nuclear installations – at NPP Mochovce and also at the plant for Decommissioning of nuclear installations, radwaste and spent fuel management in Bohunice.

The role of these committees is to monitor the global development in the area of safety of nuclear installations and to propose strategic procedures and solutions for application at nuclear installations of SE, in compliance with the European standard and safety standards of the IAEA in Vienna. It also regularly assesses the status of nuclear safety of nuclear installations of SE and proposes necessary measures for permanent sustainability and improvement of its level.



NSC - Nuclear Safety Committee
TC - Technical Committee

9. When will the advanced emergency operational procedures (EOPs), e.g., symptom based EOPs, be implemented at nuclear power plants? **[Austria]**

Answer:
Project of EOP development and implementation at V-2 units was split into following phases:

- Emergency Operating Guidelines Development
- Transformation Emergency Operating Guidelines to Procedures
- EOP Verification
- EOP Validation
- Personnel Training

Phases of EOP development , verification, validation on Full Scope Simulator (FSS) were completed. The guidelines, which were not validated yet, because of limitation of FSS simulation, would be validated during second phase of validation on Multifunction Simulator (MFS) in May 1999 in Dukovany. Process of personnel training started in1996 and will be finished in June 1999. After validation on MFS the EOP licensing and approval process will be started so that EOP should be implemented on 3-rd an 4-th unit of Bohunice NPP after refueling in Unit No. 3 in September 1999.

Symptom based Function Restoration Procedures have been implemented at unit 1 and 2 in 1993.

Symptom based EOPs will be implemented at NPP Mochovce on the 1. Unit from the 2. fuel cycle, on the Unit 2. from the commissioning.

10. In 1998 the new operational limits and conditions for the Unit V-1 of Bohunice NPP were introduced. (in the format of the Westinghouse firm). What are the differences from the ones acting before? **[Ukr.]**

Answer:

In 1998 only some parts of operational limits and condition were modified which are related to the gradual reconstruction of V-1 units. During this reconstruction important modification of safety systems, reactor protection system and safety related systems was done. According to SAR for „Gradual reconstruction„ it was necessary to modify also related parts of operational limits an conditions. There were no changes in the format of OLC. The Westinghouse format of OLC is in use in Bohunice NPP since 1984 at V-2 and since 1988 at V-1 plant. Original OLC were based on Soviet „Technological Reglament„. **(Ref. 19/6, 19/7, 19/12)**

11. Which measures are applied to address the effects of ageing phenomena on the safety of the plant? **[GERMANY]**

Answer:
The impact of construction materials ageing on the operational safety of EBO is monitored on several levels. Assessment of the equipment life which is safety important is done continuously on individual units after the end of each fuel cycle. Assessment covers the most loaded equipment including RPV and components of the primary and secondary circuit.
Assessment of equipment lifetime expiration is made from the aspect of highly deformational (low-cycle) fatigue, which (based on international experience) is the limiting degrading mechanism. Corrosion-erosion damaging mechanism is negligible for materials of austenitic type (despite that the primary circuit includes so called corrosion loop which continuously monitors the status of the corrosion damage during operation). Carbon and ferritic steel (pressurizer, SG, stem and feed water piping) are monitored periodicaly using the method of measuring the loss in the wall thickness and monitoring defects at potentially critical points. All equipment which is safety important is thoroughly re-qualified.

12. New OLC (Operation Limits and Conditions) for Bohunice V-2 were issued in March 1998 subject to the approval by UJD; they have been split into two separate documents (Unit 3 and Unit 4). Which measures or modifications made this necessary? **[GERMANY]**

Answer:
The plant policy for operational documentation is to have unit specific document for each activity in the unit. Specific unit documents are color-coded, i.e. procedures are printed on unit specific color paper. The reason for this measures is to reduce potential confusion of personnel when using procedure for certain unit.
The new OLC are practically identical for Unit 3 and Unit 4. The differences are only in numbering of equipment which is unit specific. We experienced significant benefit of this approach at V-1 plant during gradual reconstruction, because each unit was at different stage of implementation of measures. There was a need to reflect this fact in OLC. **(Ref. 19/10, 19/6, 19/7)**

13. What kind of on-site accident management measures are under consideration or already in place to prevent severe accidents or mitigate their consequences (e.g.

containment venting, bleed&feed, additional emergency power supply)? **[GERMANY]**

Answer:

There are several possibilities for accident management measures. Most important of them are listed bellow:

- Feed & Bleed possibility on the primary side,
- supply feed water to SG by a mobile diesel driven pump (there are nozzles installed on emergency feed-water lines) which is available in the plant (fire truck),
- power supply from the near hydro plant (hydro plant is equipped by DG to be able to start in case of loss of grid) by direct line from the hydro plant to emergency power bussbars at Bohunice plant,
- supply important components by independent cables (led outside cable channels trough the ground) from hydro plant or from Bohunice V2 DGs.
- post accident containment venting by using of the newly installed system
- boron injection and feedwater supply from neighboring unit,
- mobile dieselgenerator to supply 0,4 kV bussbars

EOPs prepared by Westinghouse for EMO include also beyond-design conditions not leading to melting of core and cover also the feed +bleed procedure, as well as total black-out event. Procedures for conditions leading to melting of the core will be developed from 1999.

14. Is there feedback and exchange of information on operational experience with comparable plants? **[GERMANY]**

Answer:

In the area of information exchange Bohunice and Mochovce NPP have close contacts with Dukovany NPP (Czech Republic) and Paks NPP (Hungary). In frame of WWER Club a special group, focused on operational experience feedback exchange, is working on regular bases.

In addition, both NPP are dealing with nuclear industry information from WANO network, which contains all reportable events within the world. SE, a.s. is a member of WANO Moscow Center. Our interest is focused on WWER 440 operational events. After initial screening the information are distributed to the plant personnel, analyzed and if applicable, necessary corrective action are submitted and implemented.

The Operational Experience feedback group at both sites have close contact with relevant personnel in above mentioned plants as well. The results of the plants in the area of operational experiences application are regularly compared and the effectiveness of process is evaluated.