4. Safety Analysis Reports for WWER 440 reactors

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4.1 Introduction

Implementation of nuclear power program in each country is connected with establishment of the regulatory body for safe regulation of the siting, construction, operation and decommissioning of nuclear installations. Licensing, one of the most important regulatory surveillance activity is based on independent regulatory review and assessment of information on nuclear safety of each particular nuclear installation. Documents which are required to be submitted to the regulatory body by the applicant (licensee) in Slovakia for this review and assessment, usually known as Safety Analysis Report (SAR) are presented in this report.

4.2 History of legal basis concerning SAR

Presentation of safety relevant information to the regulatory body, to support licensing applications, in a form of comprehensive document —SAR was legally established in former Czechoslovakia in 1976 when Civil Construction Act (50/76) was issued. In related regulations No. 83/76 issued in the same year general requirements on contents of safety analysis report were defined . Based on requirements of mentioned act and regulation, the applicant for construction of a nuclear facility had to submit to the regulatory authority three types of SARs. The first one, the Introductory SAR, was a part of application for the site approval. It included a brief description of the main design facility features and estimated facility environmental impact. A Preliminary SAR was elaborated and reviewed before the construction permit could be issued. It described and provided analytical and experimental evidence that all requirements on nuclear safety defined by the Introductory SAR were fulfilled in the design. QA programmes for components manufacturing and NPP construction were also parts of the PSAR. The third type of SAR was a Pre-operational SAR. It provides evidence of installation`s nuclear safety, as well as description of design changes made during construction, before first fuel loading.

Detailed content and format of safety analysis report for individual nuclear installation was determined case-by-case in a form of agreement between regulatory authority and applicant. Technical document „Rules for elaboration and guideline on content of safety analysis reports“ which was approved and issued by Czechoslovak Atomic Energy Commission (CSKAE) was used as a background document for such agreements. This document was innovated in 1977. As usual content of SAR described in the above given guideline was extended by some new chapters, representing recent development in the area.

In accordance with the above mentioned procedures, safety analysis reports were elaborated and based on review and assessment results and start of commissioning was approved by CSKAE (for V-1 Units in 1978 and 1980 and for V-2 Units in 1984 and 1985).

In 1984, a new Act on State Supervision of Nuclear Installations was issued (No. 28/84) in former Czechoslovakia. In this Act the responsibilities of state supervisory body on nuclear safety were defined more precisely. The legal duty of licensee to provide information to the regulatory body to support regulatory review and assessment was also explicitly included in this act. Also, safety important modifications on nuclear installations were stipulated for regulatory review and approval in this act.

In 1988 CSKAE guidelines „standard content of technical justification of NPP safety – safety analysis report" was issued. This guideline was elaborated in accordance with the former COMECON standards.

In 1993 all existing legal bases in former Czechoslovakia were kept in force in Slovakia, including those nuclear regulations and decisions made by CSKAE.

In 1993 UJD approved the use of US NRC RG 1.70 (Standard Format and Content of Safety Analysis Report), adopted on country specific conditions for elaboration of innovated SAR for Bohunice V-2 units. This innovated SAR was elaborated after 10 years of NPP operation.

In 1998 a new Act on Peaceful Use of Nuclear Energy was issued in Slovakia. Safety documentation for each licensing step (i.e. siting, construction, operation, decommissioning) is clearly defined. This shall be submitted to regulatory body to support review and assessment during the licensing process. Two sub-set of safety documentation are distinguished in this Act and regulatory involvement was the basis to create them. One sub-set are the most important regulatory documents which are finally approved by the regulatory body (Limits and Conditions, Commissioning programmes, QA programmes, On-site emergency preparedness plan). Another sub-set documents are required to provide information for regulatory body needed to understand the nature of the plant and safety evaluations performed to demonstrate the design safety as well as operational safety (i.e. safety analysis report, plan of physical protection, radwaste and spent fuel management system, decommissioning plan, in-service inspection programme, surveillance programmes, most important operating procedures). Second sub-set of documents are reviewed and assessed by regulatory body, but they are not approved.

There is an obligation in the authorisation act for the regulatory body of Slovakia to issue regulation(s) where contents of each above mentioned safety document will be determined. Usual content of safety analysis report will be reduced, taking into account that some of its original chapters are independent documents.

4.3 Current status of SAR for individual nuclear installations

Overall nine nuclear installations (according the definition in „Atomic Act" 130/87) are disposed in the territory of Slovakia.

According to their purpose they can be assigned to following types:

- nuclear power plants (4), in construction, operation and decommissioning phases
- interim spent fuel storage
- near surface radwaste disposal facility
- radwaste treatment facilities (3).

Current content and format of SARs as well as acceptance criteria are gradually „tailored" on above mentioned types of nuclear installations.

Overview of development process of SARs for individual nuclear installations is given in the following table:

BOHUNICE V-1 Units

		Date	*Developed*	*Contents and Format*
A	Introductory SAR	1972	TEPLOENERGOPROEKT LENINGRAD	Russian Standards for TOB
B	Preliminary SAR	1978	EBO	CSKAE Guidelines 1977
C	Pre-operational SAR			
D	Innovation of SAR after 10 years of operation			
	- innovation of accident analysis chapter	1990	VUJE	--
E	Safety upgrading			
	- innovation of accident analysis chapter after small reconstruction (new PIEs)	1993	VUJE	--
G	Major safety upgrading			
	- Introductory SAR for Major Safety upgrading	1992	VUJE + EGP	CSKAE Guideline 5/88
	- Innovated introductory SAR for gradual safety upgrading	1993	VUJE + WESTINGHOUSE + EGP	CSKAE Guideline 5/88
	- Preliminary SAR for systems modified	1994-1999	VUJE + SIEMENS + VUEZ Tlmace	
	- Completion of new SAR required as a final step of safety upgrading programme	2000	VUJE + SIEMENS	adopted US NRC RG 1.70

BOHUNICE V-2 Units

	Date	Developed	Contents and Format
A Introductory SAR	1974	TEPLOENERGOPROEKT LENINGRAD	Russian Standards for TOB
B Preliminary SAR	1978	EGP Prague	CSKAE Guidelines 1977
- amended	1979	EGP Prague	CSKAE Guidelines 1977
C Pre-operational SAR	1983	EBO	CSKAE Guidelines 1977
PERIODIC SAFETY REVIEW (CSKAE Decision No. 199/91)			
D Operational SAR	1995	VUJE	adopted US NRC RG 1.70
- amended	1998	VUJE	adopted US NRC RG 1.70

MOCHOVCE NPP

	Date	Developed	Contents and Format
A Introductory SAR	1980	EGP Prague	CSKAE Guidelines 1977
B Preliminary SAR	1984	EGP Prague	CSKAE Guidelines 1977
- amended	1986	EGP Prague	CSKAE Guidelines 1977
C Pre-operational SAR	1983	EBO	CSKAE Guidelines 1977
1st revision	1989	SKODA Prague	CSKAE Guidelines 1977
2nd revision	1997	SKODA Prague, EGP, EUCOM, VUJE	adopted US NRC RG 1.70
- amended (required)	1999	SKODA Prague, EGP, EUCOM, VUJE	adopted US NRC RG 1.70

		Date	Developed	Contents and Format
BOHUNICE A-1 NPP				
(DECOMMISSIONING PHASE)				
A Preliminary decommissioning plan (1st phase)		1992	EGP Prague	--
B Environmental impact assessment				
- Decommissioning plan (1st phase)		1994	DECOM Slovakia	IAEA draft SS 111-S-6 NRC-DG-1005 adopted for Slovakia
- amended		1995	DECOM Slovakia	IAEA draft SS 111-S-6 NRC-DG-1005 adopted for Slovakia
- Safety Analysis Report for plant current status		1996	DECOM Slovakia	NRC RG 1.70 adopted for decommissioning
C Conceptual decommissioning plan (all phases)		1998	DECOM Slovakia	IAEA SS111-S-6 draft SG G.6.1
D Small radwaste treatment and conditioning technologies: vitrification (I), fragmentation (II), decontamination (III)				
- Preliminary SAR	(I)	1987	EGP Prague	--
	(II)	1995	EKOSUR	US NRC RG 1.70 adopted for waste management facility
	(III)	--	--	
- Pre-operational SAR	(I)	1995	EBO	US NRC RG 1.70 adopted for waste management facility
	(II)	1998	EKOSUR	US NRC RG 1.70 adopted for waste management facility
	(III)	1998	AllDECO	US NRC RG 1.70 adopted for waste management facility

		Date	Developed	Contents and Format

BOHUNICE INTERIM SPENT FUEL STORAGE

		Date	Developed	Contents and Format
A	Introductory SAR	1982	EGP Prague	CSKAE Guidelines 1977
B	Preliminary SAR	1982	EGP Prague	CSKAE Guidelines 1977
C	Pre-operational SAR	1985	SKODA, EBO	CSKAE Guidelines 1977
SAFETY UPGRADING and STORAGE CAPACITY EXTENSION				
a)	Environmental impact assessment			
	- EIA report	1995	VUJE	Act No. 127/94
	- amended	1996	VUJE	Act No. 127/94
b)	Preliminary SAR for safety upgrading	1996 (1st ver.) 1997 (2nd version)	VUJE	US NRC RG No. 3.44 adopted for Slovakia

MOCHOVCE SURFACE DISPOSAL FACILITY

(Defined as Nuclear Installation in 1987 by CSKAE regulation 67/87)

		Date	Developed	Contents and Format
A	Preliminary SAR	1984	UJV Řež CHEMOPROJEKT Prague	CSKAE Guidelines 1977
B	Pre-operational SAR			
	1st revision	1993	Mochovce NPP + SCK Belgatom	NUREG G1199 adopted for Slovakia
	2nd revision	1998	VUJE	NUREG G1199 adopted for Slovakia

BOHUNICE CONDITIONING CENTRE

		Date	Developed	Contents and Format
A	Preliminary SAR	1993	EGP Invest + DECOM	adopted according RG 1.70
B	Pre-operational SAR	1998	SLOVRIA	adopted according RG 1.70

4.4 Status of plant specific PSAs

A major effort is under way at the present time in the countries operating WWER type NPPs to establish dependable PSAs to be used for balancing decisions on backfittings and to enhance the safety of these plants. Since its constituting in I993, Nuclear Regulatory Authority of the Slovak Republic has devoted an effort to use PSA technology to support the regulatory policy in the Slovak Republic. PSA technology for NPPs has been becoming a standard tool to further enhance the safety of these installations in Slovakia.

PSA of the Bohunice V-1 Units was performed by UK company Electrowatt in co-operation with national engineering companies RELKO and VÚJE Trnava. During the course of the work that was financed by PHARE program, two IAEA international peer review service (IPERS) missions reviewed quality of the study. PSA of Bohunice V-2 plant was undertaken by the national companies mentioned above. This PSA model was also subject of an IAEA review mission. Several proposals for V-1 and V-2 Units upgrading measures based on the PSA studies have been implemented. Reasons to incorporate risk-based regulatory approach into the policy of ÚJD SR have found their practical application.

4.4.1 Bohunice V-1 Units PSA

Some PSA results

The scope of the PSA Level 1 study has covered only full power mode, considering all important initiating events, including internal fires and floods. The study has been carried out for three different states of the plant - prior to the small reconstruction, after the small reconstruction and a state with some measures of the gradual reconstruction. Objectives of the study were:

- to estimate the core damage frequency (CDF)
- to identify the most significant accident sequences in terms of CDF
- to assess the influence of the plant modifications implemented on CDF
- to indicate recommendations for updating some emergency management procedures and technical specifications.

The calculated CDF for the state prior to the small reconstruction was 1.7 x 10^{-3}/reactor- year with rounded off contributions of dominant initiating events to CDF as follows:

internal fires	5.6 x 10^{-4}	33 %
LOCAs	5.1 x 10^{-4}	30 %
last turbine generator (TG) trip	3.1 x 10^{-4}	18 %
pressurizer steam LOCA	2.0 x 10^{-4}	12 %
loss of main feedwater (MFW)	9.7 x 10^{-5}	6 %
TOTAL	1.7 x 10^{-3}	99 %

The calculated CDF for the state after the small reconstruction was 8.89 x 10^{-4}/reactor-year with rounded off contributions of dominant initiating event categories to CDF as follows:

last TG trip	3.4 x 10^{-4}	38.2 %
pressurizer steam LOCA	1.3 x 10^{-4}	14.8 %
medium LOCA (32 -100 mm)	8.6 x 10^{-5}	9.7 %
very small LOCA (< 7 mm)	8.1 x 10^{-5}	9.1 %
SG tube rupture (1 tube)	4.8 x 10^{-5}	5.4 %

multiple SG tube rupture	4.7×10^{-5}	5.3 %
small LOCA (7 mm - 32 mm)	4.2×10^{-5}	4.7 %
loss of MFW (total)	2.4×10^{-5}	2.7 %
partial loss of MFW (< 3 pumps)	1.8×10^{-5}	2.0 %
steam header break	1.1×10^{-5}	1.3 %
TOTAL	8.3×10^{-4}	93.2 %.

Based on the above results, the following priorities for additional plant safety upgrading, i.e. for the gradual reconstruction have been proposed in order to:

- to increase reliability of the high pressure safety injection system
- to provide additional analysis regarding required number of spray coolers needed to
- maintain temperature in emergency water tank in post LOCA re-circulation mode
- to change manually operated valves on spray coolers by electrical ones
- to provide analysis of the impact of last TG trip signal failure on the plant safety
- to increase reliability of TG protection in order to decrease initiating event frequency
- to increase reliability of reactor protection system in case of overpressure transients
- to develop symptom based emergency procedures.

It is necessary to emphasize that available PSA results are, of course, not the only criteria for measures of the gradual reconstruction, which are much broader. Basic engineering for implementation of those measures has been developed by German company Siemens. Impact of the basic engineering measures was evaluated by PSA and some preliminary results have shown the potential for decreasing CDF below the value of 1×10^{-4} on condition that symptom based emergency operating procedures will be developed and operators will be trained to use them. At present, all proposed modifications are being evaluated by PSA model and recommendations resulted are submitted to design organizations for further implementation.

4.4.2 Bohunice V-2 Units PSA

Some PSA results

The scope of the PSA Level 1 study has covered only full power mode, considering all important initiating events, including internal fires and floods. The study has been undertaken for two different states of the plant. Objectives of the study were to estimate the CDF and to determinate dominant contributors to risk. The first PSA model was developed for the virtually original design and the model identified that the emergency power supply system (so called category 2) had dominant contribution to CDF. It was shown that following some specific signals (large and medium LOCA, steam header break), the normal power supply to the 6 kV emergency busses was automatically disconnected, which resulted in starting dieselgenerators and automatic loading without real loss of power supplying. That was why some design modifications were proposed to increase the reliability of the power supply. The second PSA model was developed after implementation of these modifications, covering also recovery actions and common mode failure in the turbine hall. The plant specific data were used for all safety system pumps, dieselgenerators, valves and for some parts of I&C systems. Calculated value of CDF, corresponding to this model, i.e. to the present state of the plant is 6.41×10^{-4} /reactor-year with contributions of dominant initiating events to CDF as follows:

loss of off-site power	1.9×10^{-4}	30.5 %
steam line break outside the containment	1.2×10^{-4}	19.0 %
main steam header break	9.3×10^{-5}	14.5 %

fire in TG hall	6.9×10^{-5}	10.8 %
feedwater header break	4.9×10^{-5}	7.7 %
interfacing LOCA	3.2×10^{-5}	5.1 %
very small LOCA (< 7 mm)	3.1×10^{-5}	4.9 %
medium LOCA (20 - 40 mm)	2.2×10^{-5}	3.4 %
pressurizer steam LOCA	8.3×10^{-6}	1.3 %
SG tube rupture	5.1×10^{-6}	0.8 %
TOTAL	6.3×10^{-4}	98 %.

Based on the PSA results, the safety upgrading measures as follows have been proposed to be implemented:

- to change electric power supply to valves in demineralized water system and to change signals for automatic opening the valves in the event of loss of off-site power
- to change electric power supply to valves of emergency feedwater (EFW) system and to
- change normal operating position to "open" for isolation valves of the EFW header
- to develop symptom based procedures for feed and bleed initiating failure and some other
- failures considered in the assessment
- to improve fire protection in the TG hall.

4.4.3 MOCHOVCE NPP PSA

Status of PSA

Within a project financed by the utility the level 1 PSA study of Unit 1 of Mochovce NPP was developed by VÚJE Inc. and RELKO Ltd. The study has two phases where the pre- and post-modification state of the plant is evaluated. The pre-modification state is the plant state before implementation, the post-modification state is the plant state after implementation of the safety measures which were specified in safety enhancement program. In April 1999, first phase was completed, second phase is expected to be completed in 2000 year.

PSA report contains a description of methodology and results obtained for the pre-modification state of the study are summarised. The objectives of the pre-modification part of study are as follows:

- Estimation of the core melt frequency using fault/event tree methodology
- Identification of initiating events and dominant accident sequences with the highest contribution to the core melt frequency
- Identification of the possibilities for operational safety improvement
- Preparing input for comparison with the post-modification state PSA results to evaluate the benefit of safety measures from the risk point of view.

The initiating events under consideration were those internal plant initiators that could lead in combination with the safety system failures to the core damage. The study includes LOCAs, transients, internal fires and floods on 100% power operation. External events, as earthquake, aircraft crash, etc. are going to be included only into the post-modification model.

The accident sequences have been modelled using event trees, where the consequences have been identified in dependence on the success or failure of safety systems. The consequences regards the core damage.

The reliability of the front-line and support systems was calculated using fault tree methodology. Component failures, common mode failures and pre- and post-accident human errors have been considered in the analysis. 24 hours mission time is used in the evaluation of the post-accident reliability of the systems.

The small event tree and large fault tree approach has been used. The model has been developed in the RISK SPECTRUM PSA code.

The study has been prepared in accordance with the IAEA procedure for conducting level 1 PSA of NPP. The following activities are included:

- Document Collection and Plant Familiarisation
- Initiating Event, Accident Sequence Analysis, Success Criteria
- System Analysis
- Data Analysis
- Human Reliability Analysis
- Internal Fire Analysis
- Internal Flood Analysis
- Core Damage Frequency Quantification and Interpretation of the Results.

PSA study is now in reviewing and assessment process.

4.5 Policy of the regulatory authority

All NPPs in Slovakia have been designed on a deterministic basis. For all the units this has been formalized by a decision (made when the plants were ordered) to apply the rules governed by the former Soviet nuclear regulatory authority and CSKAE, including several internationally recognized and applied safety standards. Based on national as well as international operating experience and indications resulted from PSAs, ÚJD SR since its constituting in l993 has devoted an effort to use PSA technology to support the regulatory policy in Slovakia. It has been judged useful by the utility, by the architect engineer, the engineering companies and by the research/development institutes involved to carry out a PSA study in the framework of the periodic safety review, as is now common practice in Europe. The PSA is considered as a complement, not as a substitute, to the deterministic approach. A combination of deterministic approach (comparison of a current status with prescribed quantitative or qualitative targets, e.g. type of systems, redundancy, separation, ranking of safety issues into categories according to their impact to safety) and probabilistic approach (calculation of the overall NPP risk expressed e.g. by CDF, identification of dominant initiating events, systems, failures according to their contribution to risk) is required.

Suchlike combined approach is used in decision making processes and the final decision on scope and priorities is based on it. There is a strong support to use PSA methods for priorization of upgrading measures, analyses of individual contributions to risk reduction and for comparison of interim indicative safety targets, values of them are still under discussion at ÚJD SR. Nevertheless, in accordance with current common practice accepted in most PSAs of proved reactor designs, the following values can be given: CDF less than 10^{-4} per reactor-year, reactor scram failure probability less than 10^{-5} and screening criterion for external events less than 10^{-7} per reactor-year. The following reasons should be mentioned as to incorporate risk-based regulatory approach into decision making on safety issues in Slovakia:

- effective evaluation of alternative safety upgrading measures
- checking safety level in comparison with other NPPs in operation
- effective improvement of plant safety with limited resources

- potential for comparison of safety of three NPP types (V-1 and V-2 under operation, the third plant of advanced WWER-440/V-213 design is under construction at Mochovce site) in Slovakia with many differences in design and age
- potential for quantification of interim safety goals in upgrading processes
- demonstration of improvements in plant safety due to extensive plant modifications
- coherence and consistency with decision to follow the IAEA recommendations on core melt and radioactivity releases criteria (PSA Level 2 efforts on V-1 and V-2 plants have recently begun).

For UJD SR, PSA of NPP was noncompulsory in the past, but at present it is a tool to better optimize safety since it helps to rationalize decisions on "how much safety is enough safety". Therefore, it is required as a part of safety documentation. Independent IAEA peer review of PSA is a common practice. For plants in operation, PSA results are used mainly for proritization of safety upgrading measures. For Bohunice V-1 plant for example, the operation permit is issued only for one year period and after each stage of upgrading, probabilistic assessment is required to indicate anticipated impact on CDF.

4.6 International assistance received

Safety analysis reports given in tables in chapter 3 are reflecting safety enhancement process, mainly for nuclear power plants. Development of the safety enhancement programmes for individual NPPs in Slovakia was tightly connected with IAEA activities, particularly with the Extra-budgetary programme launched in 1990.

Reasonable contribution to the development of safety enhancement programmes for NPPs in Slovakia were IAEA activities oriented on identification of safety issues as well as their ranking in accordance with their safety importance. Identified safety issues in parallel to results of national design safety re-evaluation and results of operational experience were used for development of conceptual programmes for safety enhancement.

For reviewing and assessment of conceptual safety upgrading programmes regulatory body of Slovakia requested IAEA to support UJD at decision making processes and by conducting specific safety review missions. One of the first mission of this type in Slovakia was invited to Bohunice V-1 NPP for review of Bohunice V-1 Major upgrading programme (Piešťany, July 1993). In the same year another safety mission of this type was invited to Mochovce NPP to review Safety Improvement Programme for Mochovce NPP (December 1993). This safety improvement programme was elaborated in co-operation of NPP Mochovce and EdF. Another mission of this type, with a limited scope was invited also to Bohunice V-2 Units (September 1984). This mission was oriented on accident analysis results elaborated as the innovation of SAR after 10 years of operation.

Following design studies during the design development were supported by IAEA activities in specific areas, like:

- leak before break concept
- RPV embrittlement and annealing
- fine hazard analysis
- probabilistic safety assessment
- confinement evaluation
    - ⇒ bubble condenser metalic structure
    - ⇒ leak rate measurement

    - ⇒ confinement improvement options for WWER 440/V230 type
- qualification of in-service inspection systems
- I&C for WWER 440/V230
- horizontal steam generators
- seismic evaluation
- guidelines for
    - ⇒ accident analysis
    - ⇒ PTS analysis
    - ⇒ shut-down conditions
    - ⇒ primary to secondary leaks
    - ⇒ ATWS analysis.

Documents elaborated in mentioned areas were used either by operators (and their suppliers) or by the regulatory body.

Additional IAEA missions were invited to help the regulatory body of Slovakia for reviewing and assessment of specific safety upgrading measures before their implementation, like:

- mission to Bohunice V-1 Units for reviewing of LBB concept
- review of Mochovce NPP bubble condenser metalic structure stress conditions
- IPERS missions to Bohunice V-1 and V-2 Units.

Finally, IAEA missions were invited to Slovakia to review how the IAEA recommendations obtained mainly in IAEA-EBP safety issue books were implemented. Such mission was held at Bohunice V-1 Units in June 1998 and at Mochovce NPP in October 1998 (financed from RER Programmes).

When concluding the IAEA Extra-budgetary Programme, an important feature of this programme was recognised, i.e. due to the co-operation of high number of experts from variety of countries, international consensus on the safety issues related to WWER reactors was reached finally.

Another, very important source of assistance, directly influencing safety re-assessment and development of safety enhancement programmes was assistance of European Union. Activities of EC under the PHARE Programme in Slovakia were oriented on particular safety issues like:

- review of I&C systems availability on WWER 440/V213 reactor type (1993)
- evaluation of WWER 440/V230 reactor type confinement improvement options
- review of safety improvement programme for Mochovce NPP (independent safety review made by RISKAUDIT – 1994)
- qualification of bubble condenser for WWER 440/V213 reactor type
- assistance to the regulatory authority of Slovakia at the licensing of Bohunice and Mochovce NPPs

International assistance provided for Slovakia in period since 1990 till now as well as assistance agreed on bilateral level with USA, U.K., Switzerland, Canada, Japan, France and Germany contributed very essentially to the development of safety upgrading programmes and to their high quality in Slovakia. At the same time internal assistance, as well as bilateral assistance essentially supported strengthening of the regulatory body and establishment of enhanced regulatory regime in the Slovakia.

5. Annexes

5.1 List of international missions on NPPs in Slovakia

5.1.1 Safety Assessment of Bohunice V-1 Units

The following safety review missions have been conducted during the operation of the Bohunice V-1 units:

- IAEA Fact Finding Mission, Sept. 3 - 7, 1990; the objective of the Mission was to document safety measures taken to improve safety of operation of units 1 and 2 throughout their operation time: operation control, facility improvements, staff training, and control and testing activities.
- Mission of the Siemens company to assess the project and safety level, August - November, 1990. Based on deterministic assessment of the project and safety level, the group of experts provided recommendations for safety improvements; they were accepted by the operator, and included in the „Small Reconstruction" program.
- Commission of the CSFR government and Federal Ministry of the Environment - August - September, 1990. The aim was to asses the current status of nuclear safety of V-1 units and to review the impacts and possibilities to resolve the energy situation of Czechoslovakia if a need should occur to immediately shut down V-1 units because of insufficient safety. Based on the report, CSKAE conducted a comprehensive assessment of V-1 units current condition. The reports clearly stated that there was no need to immediately shut down the power plant, in spite of certain deficiencies, in particular concerning V-1 design. CSKAE issued the Resolution No.5/91 to change approvals on permanent operation of both units, issued by CSKAE in 1980 and 1981, and it regulated the further operation of the units. The operation of the units is now subject to annual approvals based on the progress of upgrading. In addition, the necessary safety improvements to be implemented were defined.
- Austrian Expert Commission, August - October, 1990. The objective of the Commission's visit was to collect information on V-1 unit's safety and to recommend to the government of the Republic of Austria how to further proceed in negotiations with the CSFR government.
- IAEA ASSET (Assessment Of Safety Significant Events Team), October 1-12, 1990; the objective was to review the accident (operation events) prevention concept, to assess the adequacy of measures taken, and to recommend areas for improvements. All events have been reviewed which occurred since the start of operation of the units, and safety significant events were identified. Any of the events, as stated, had no radiological impact on the environment. The activities of the Breakdown Commission for Investigation of Nuclear Installations Events as well as the measures taken immediately were considered adequate. Recommendations were extended for improved process efficiency as well as for quality assurance, staff training and design improvements.
- IAEA Safety Review Mission, April 7 - 26, 1991 within the „IAEA WWER-440P230 Nuclear Power Plants Safety Program, in the framework of which the first conceptual review of these units has been performed in February, 1991. This was followed by missions to the individual power plants. The objective of the program was to review the design and the operation, considering specific conditions of the power plant, and to formulate recommendations (classified into four classes according to their safety significance) which were expected to assist in decision-making concerning the achievement of a higher safety level. The safety significant problems identified during the

missions were incorporated into the document TECDOC-640. The approach of the operator who develops its own plan to deal with the issues identified, was positively assessed as a good practice in safety culture.

- IAEA Safety Review Mission in Relation to the Design of Seismic Upgrading for Bohunice NPP, September 2 - 6, 1991; the objective of the mission was to review criteria and design documentation developed for seismic upgrading of V-1 and V-2 units. The mission appreciated the high professionality of the staff involved in the seismic upgrading project, and suggested recommendations concerning structures, systems and components the application of which would secure safe shut-down of the power plant and its maintaining in a safe condition after an earthquake.
- IAEA Safety Review Mission, April 27 - 30, 1992; the objective was to review the implementation of the recommendations and suggestions of the preceding mission (April, 1991) and to evaluate activities performed in response to the Technical Report of the mission. Also, the Report contained an evaluation concerning safety issues identified by IAEA and published in TECDOC-640 document and their degree of implementation under the „Small Reconstruction„. The Mission Report considered the activities of the power plant a satisfactory progress, and numerous safety-related issues were considered as eliminated.
- IAEA Seismic Safety Review Mission Relating to the Seismic Upgrading of Bohunice NPP, May 5 - 7,1992; the objective of the Mission was to check the implementation of the recommendations of the preceding Mission of September, 1991, to review the implementation of works relating to the seismic upgrading project of V-1 units. It was stated that the seismic risk of the V-1 power plant has been substantially reduced due to the previous upgrading works, and recommendations were extended concerning problems identified during the preceding review, and issues were pointed out accordingly.
- IAEA Seismic Safety Review Mission Relating to the Seismic Upgrading of Bohunice NPP, April 5 - 8, 1993; the objective of the Mission was to review the project and the implementation of V-1 units upgrading works as recommended by preceding IAEA Missions (September 2 - 6, 1991 and May 5 - 7, 1992) The Mission Report appreciated the volume of works done to seismically upgrade both units, and pointed to the need to continue dealing with issues identified by the preceding Missions.
- IAEA Peer Review Mission to Review the Probabilistic Assessment of V-1 Units Safety Study, March 8 - 12, 1993; the first review stage has very positively assessed the extent, organization, quality assurance, identification and grouping of initiation events, development of event trees of the Level 1 Probabilistic Safety Assessment Study, and defined certain issues to be dealt with in the next steps. The Level 1 PSA Study was developed in cooperation of the operator with the company Electorate Engineering Services, United Kingdom.
- IAEA ASSET (Assessment of Safety Significant Events Team Follow-up Mission), July 5 - 9, 1993. Its objective was to review the implementation of recommendations of the ASSET 1990 Mission and of those of the ASSET Advisory Group meeting of July, 1991, to identify the efficiency of the Conception of the prevention of power plant operation events since 1990, and to extend further recommendations to improve event prevention efficiency. The Mission Report has appreciated the energetic response to the preceding recommendations, and noticed significant progress in the safety improvements of both units. Extensive plans of safety improvement were highly commended, and continuous taking of measures has been recommended, the implementation of which not been could completed due to their significant time-consuming nature.
- IAEA „Small Reconstruction" Assessment Mission, July, 1993. The Report has stated that a significant progress had been achieved with respect to the definition and implementation of safety

improvements since TECDOC-640 was issued. At the same time the need was pointed to revise the strategy and safety implications of suggestions on the „main reconstruction program".

- IAEA Site Safety Review Mission to Review the Design Basis Seismic Input for Bohunice and Mochovce NPP Sites - October 18 - 22, 1993. The aim was to assess data and methods used in determining the impacts of design earthquake and to provide recommendations for further activities in the seismic upgrading area.
- Review of the „Leak Before Break„ Concept Application to the Bohunice WWER- 440/230 NPP, consultation meeting; February 28 - March 2, 1994. The aim was to evaluate LBB analyses conducted, consequent changes in equipment and application, as well as the general adequacy of the LBB program to meet the set safety requirements. The technical aspects and the LBB concept programs were assessed as suitable, and the results of the analyses conducted have provided evidence for meeting the criteria ;and that, some unavoidable project adjustments have been made. Operation control programs concerning the main circulation pipe are adequate.
- IAEA Peer Review Mission to evaluate PSA NPP V-1 Study, February 28 - March 11, 1994. The final evaluation continued the preceding March 1993 Mission. Two PSA models have been reviewed: PSA Level 1 - before and after „small reconstruction„ including internal fires and floods. It has been stated that the method, technique and PSA data used follow standard practices as recommended by IAEA Guidelines.
- A seminar organized by ÚJD in cooperation with IAEA to evaluate embrittlement and baking of the WWER 440 reactor pressure vessel (RPV) - March 29 - 31, 1994. The aim was to discuss issues of the integrity of WWER-440/230 reactor pressure vessels, the previously taken measures, ongoing activities and plans for the future. The evaluation report of the workshop contains recommendations with respect to RPV integrity of this reactor type.
- Consultation Meeting on Safety Improvements to WWER-440/230 NPPs) - September 26 - 30, 1994, Vienna. The meeting reviewed the previous results of the dealing with safety issues of WWER-440/230 units identified by the preceding missions, and updated the IAEA database of the status of the problem solving at the various power plants. It is evident from the assessment that among all the units, the best progress in safety improvements had been achieved at the Bohunice power plant.
- IAEA Safety Mission to Slovakia: Seismic Safety Review for Bohunice and Mochovce NPPs - October 31 - November 4, 1994. The aim was to review the tectonic stability of the subbase and to review the design parameters of earth movements for the Bohunice and Mochovce NPPs. This mission was a continuation of the October 1993 mission, and it provided recommendations for further proceeding.
- IAEA Technical Safety Review Mission - May 6 - 8, 1996 - the aim was to update information available to IAEA on the implementation of safety improvements and to review activities of the power plant with respect to safety issues resolution (both operating and design problems) identified by the preceding missions. The mission has stated that all safety measures contained in the IAEA TECDOC - 640 document concerning design and operation improvements had been reflected in the V-1 Units reconstruction program.
- IAEA Gradual Reconstruction Review Mission - June 15 - 19, 1998 - a continuation of the 1991 - 1996 Safety Review Missions. The Mission's focus was on the evaluation of the implemented and/or planned modifications of V-1 units under the gradual rehabilitation program, in particular from the aspect of the dealing with the safety significant problems of the WWER 440/230 units defined in the TECDOC 640 document. The Mission appreciated the previous approach as well as the further implementation of the safety upgrade program as far as the scope and the adequacy of the measures are concerned. The Mission also defined certain recommendations, including an

invitation for an IAEA Mission which would comprehensively assess the treatment of safety significant problems after the completion of the gradual rehabilitation program beyond 1999.

- IAEA Bohunice site Seismic Safety Review - 16-20 November 1998 concluded that RLE characteristics for the site are considered reasonable.
- WANO Peer Review - 19 October - 6 November 1998

5.1.2 Safety Assessment of Bohunice V-2 Units

In addition to the review missions mentioned above, the following V-2 units safety review missions have been visiting the Bohunice NPP:

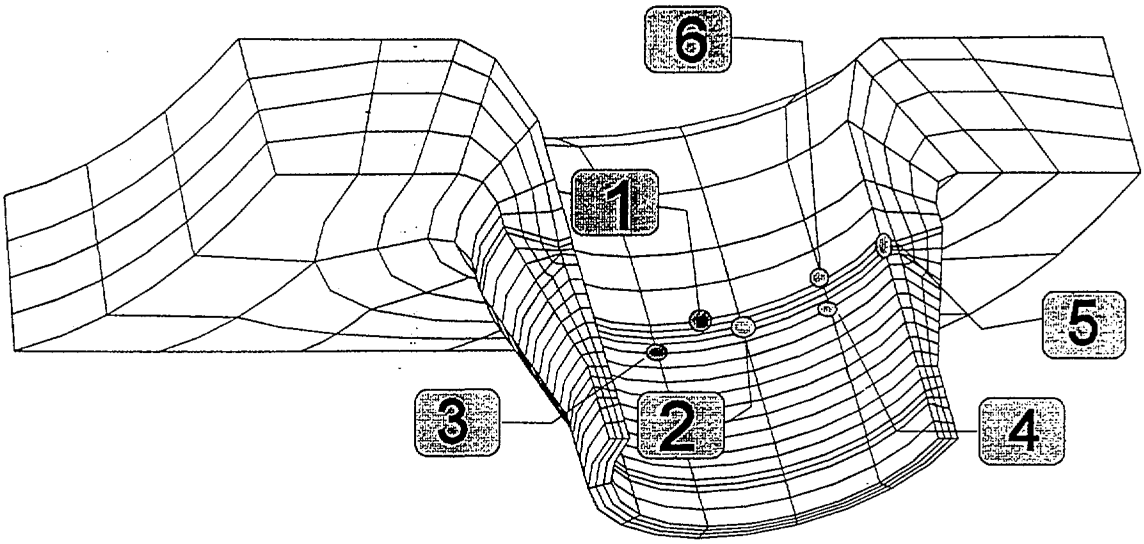
- IAEA Safety Review Mission - September 5 - 12, 1994. The aim of the Mission was to compare the NPP design with the current safety-related approach, and to provide recommendations to assist the NPP management and the regulatory body to make decisions on safety improvements. The report has stated that the NPP had made considerable endeavor to improve the original design, and provided recommendations and suggestions for further design improvements. In a next step, safety issues identified with all WWER-440/213 model power plants were put together and ranked according to their importance. Four categories were distinguished, as laid down in the material „Safety Issues and their Ranking for WWER 440 Model 213 Nuclear Power Plants".
- IAEA PSA Peer Review (Probabilistic Safety Assessment Level 1) of V-2 Units - January 17 - 28, 1995. It has been stated that the study objectives had been achieved, and all relevant initiation events with safety relevance had been considered. Effectively no modeling-related faults could be identified. The study can be a good basis for further applications in the NPP and National Nuclear Authority programs because it contains a broad range of possible applications of specific NPP data and staff expertise.
- IAEA Operation Safety Review Mission (OSART) - September 9 - 26, 1996. The aim of the Mission was to review operation procedures, to exchange experience and knowledge between and among Mission members and NPP partners on how a common objective could be pursued, namely an outstanding operation safety standard. The group offered suggestions to improve operation safety, it appreciated areas where activities are performed on a good level, and identified areas assessed as good practice to be recommended for implementation in other power plants - installation and utilization of up-to-date diagnostic systems and teledosimetric system for the monitoring of the radiation status within and in the environment of the nuclear power plant.
- Follow-up IAEA Operation Safety Review Mission (OSART - Follow-up visit) - March 2 - 6, 1998. The aim of the Mission was to evaluate activities of the NPP oriented towards operation safety-related recommendations identified by the preceding mission. The expert team stated an excellent level of preparedness of the NPP for the Mission, appreciated the willingness of the NPP management to consider new suggestions and the implementation of changes, as a positive indicator of further improvements. Of all recommendations and suggestions made by the OSART 96 Mission, 49 % had been implemented, satisfactory progress had been achieved with respect to additional 49 %, and 2 % (a single recommendation) will be solved in the near future.

5.1.3 Safety Assessment of Mochovce NPP

- IAEA Mission - for OSART, conducted on January 9 - 29, 1993, was focusing on the review of the preparedness of the operator to commission and to operate the plant. The Final Report contains recommendations for improvements in the areas of management, staff training, operation and maintenance, technical support, radiation protection, emergency planning and preparedness, as well as in the area of start-up which cover operation safety and identify good practices and activities to be considered also by other nuclear power plants.

- IAEA Mission - Safety Improvements Review for NPP Mochovce. The Mission was focusing on the check-up of safety improvements at NPP Mochovce. The aim was to discuss safety-related problems known to exist with respect to WWER 440/213 reactors, safety improvements already incorporated into the NPP Mochovce project or suggested in the Safety Improvements Report prepared by EdF and SIEMENS experts working together with Slovak organizations. The review covered the main safety functions - power control fuel cooling, maintaining of the primary circuit integrity. In addition, CMS, electric feeding, emergency analyses, internal and external risks were all considered. A report was compiled containing the findings and recommendations for individual areas reviewed, and they were incorporated into the NPP Mochovce's safety improvements program.
- IAEA Seismic Safety for Nuclear Power Plants Bohunice and Mochovce Mission. The aim of the Mission was to verify the evaluation method of seismic input data and to assess the effects of external earthquake risk on NPP safety. The prepared POSP was used as a background material. The Mission reviewed the background materials supplied, and compared them with the recommendations of the IAEA 50-SG-S1 safety guide concerning the location of NPP. In conclusion, procedures and results obtained were considered adequate.
- RISKAUDIT Mission (consortium of technical support organizations IPSN and GRS working for national nuclear authorities of France and Germany) focused on the review of safety improvements of NPP Mochovce and the assessment of design safety was conducted on December 20,1994.
- IAEA expert meeting held on 14 - 18 September, 1998 in Vienna conclude that no concerns were identified with the integrity of the Mochovce Unit Reactor Pressure Vessel.
- PHARE Licensing related assessment - since February 1998
- IAEA Mochovce Safety Improvement Review mission - 5 - 16 October, 1998 . Experts declared that all generic issues identified in IAEA-EBP-WWER-03 and IAEA-WWER-SC-102 documents have been successfully addressed by the plant. In addition the plant staff identified additional plant specific issues and has taken steps to improve equipment as necessary.
- IAEA Mochovce Seismic Safety Review - 16 -20 November, 1998 stated that seismic hazard study using the deterministic approach has been performed generally on the recommendations of the IAEA Safety Guide 50-SG-S1.

Obrázok č. 4-15 Horúci nátrubok TNR



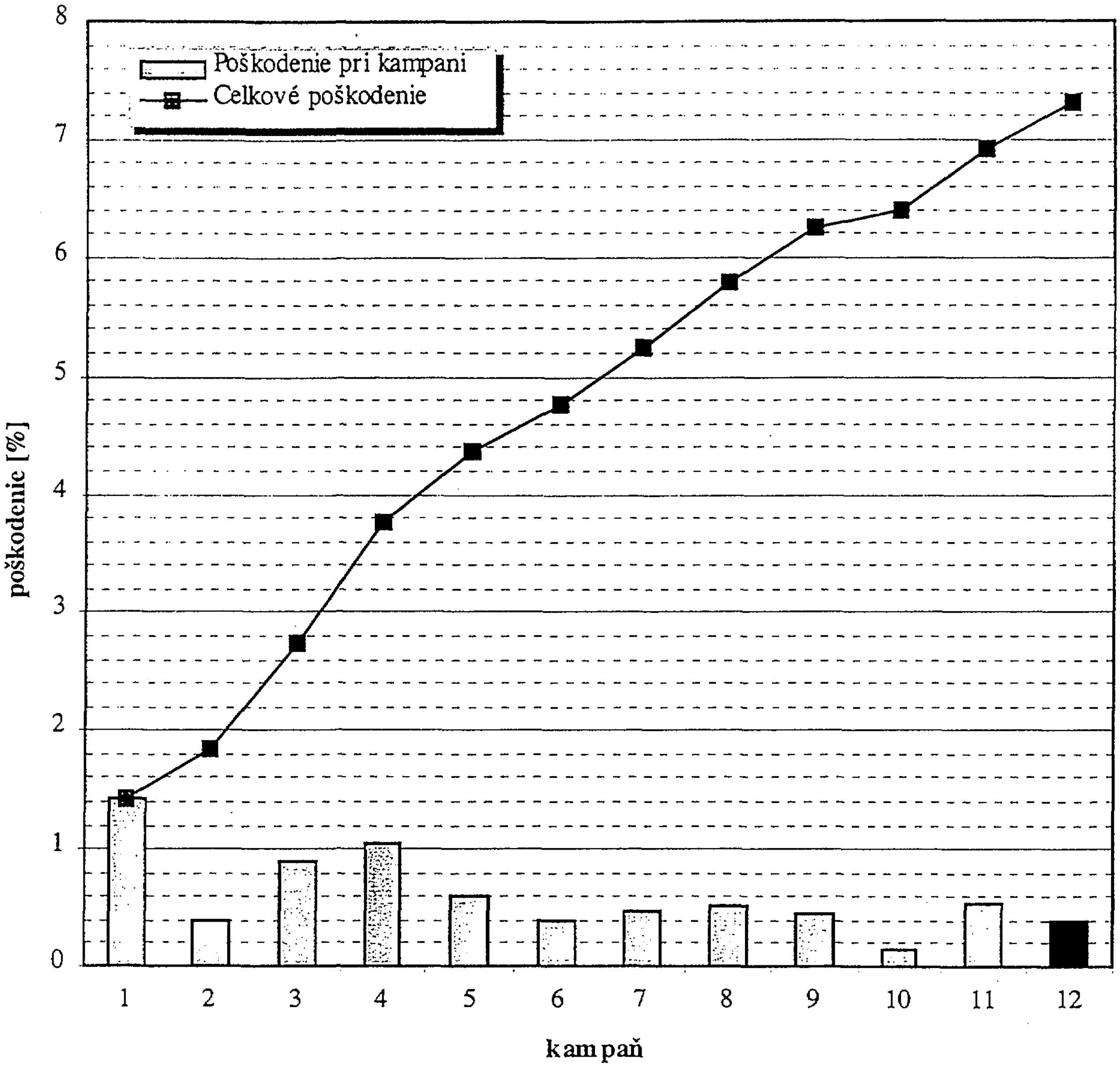
Tabuľka č. 4-15 Horúci nátrubok TNR - poškodenie pri 12 kampani

Kumulácia únavového poškodenia pri 12 kampani						
Bod číslo	1	2	3	4	5	6
Poškodenie [%]	**0,4496**	0,379	0,0251	0,0577	0,0099	0,0131

Tabuľka č. 4-16 Horúci nátrubok TNR - poškodenie po 12 kampani

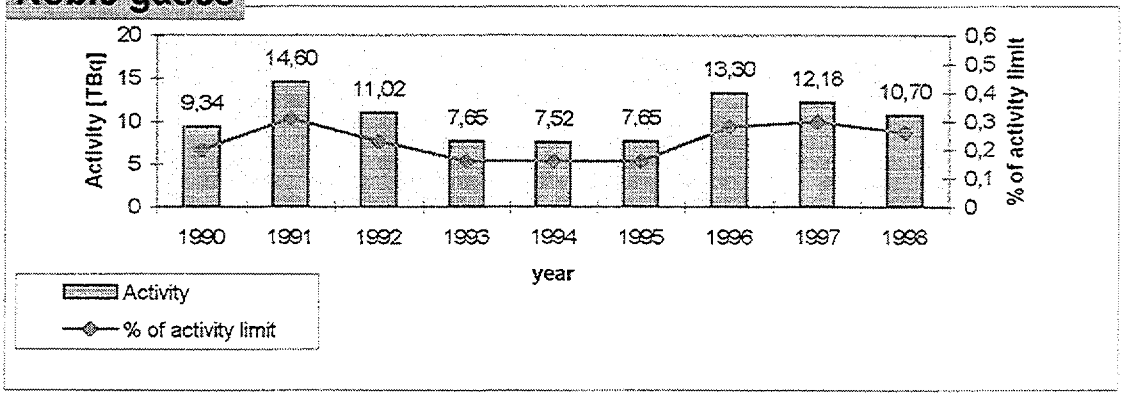
Celková kumulácia únavového poškodenia po 12 kampani						
Bod číslo	1	2	3	4	5	6
Poškodenie [%]	**7,3186**	6,3510	0,5871	1,2007	0,3349	0,1171

Obrázok č. 4-16 Horúci nátrubok TNR - graf

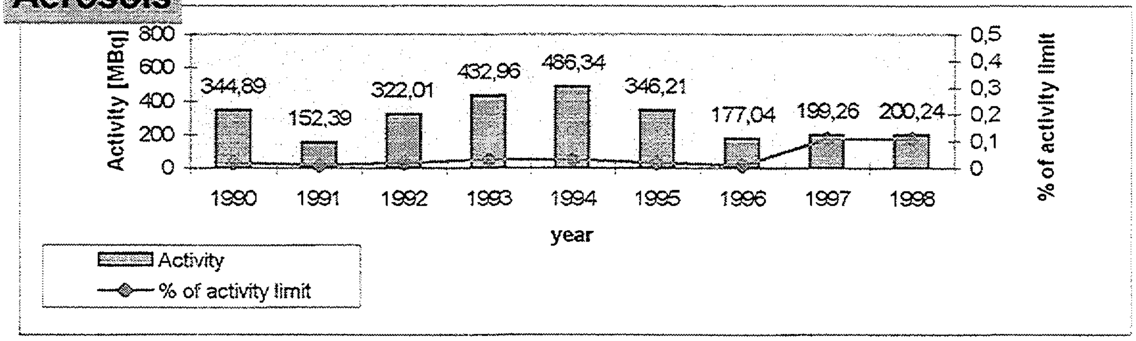


Gaseous discharges V-1

Noble gases



Aerosols



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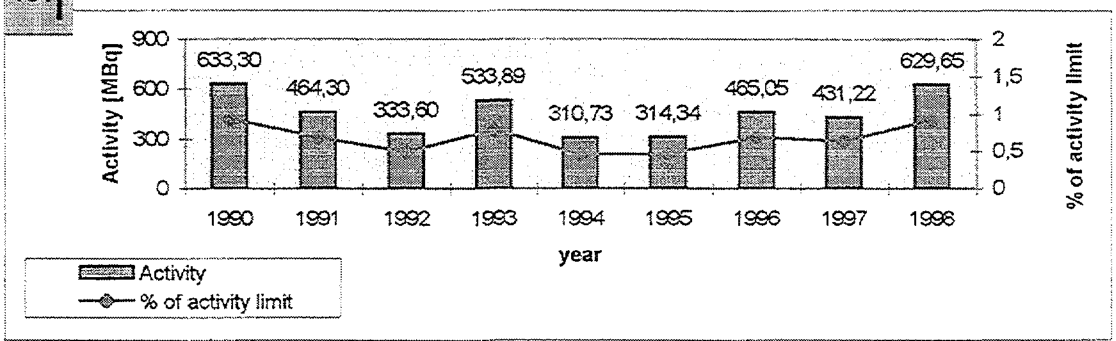

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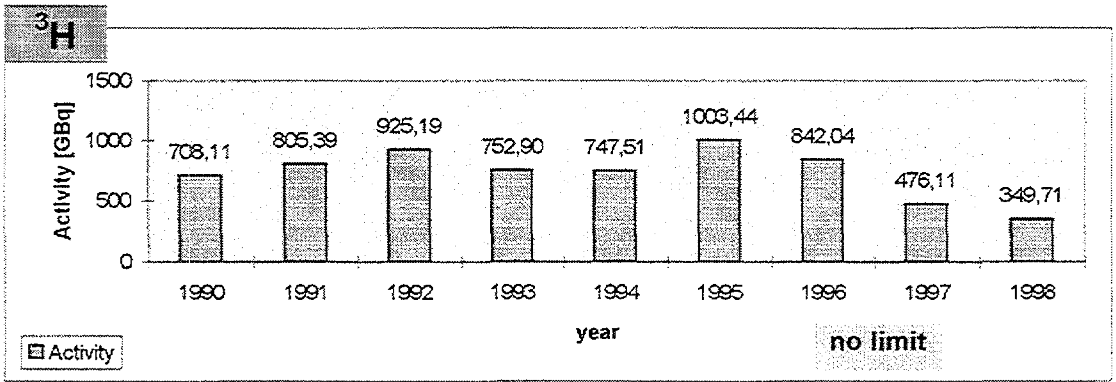
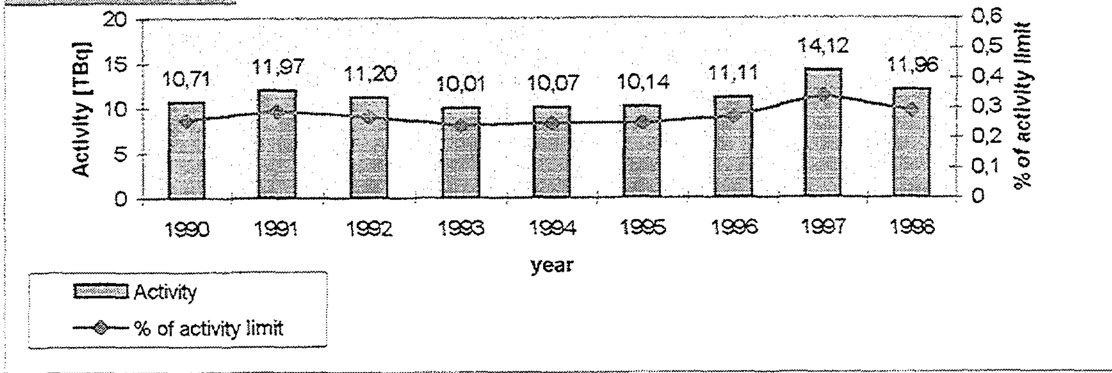
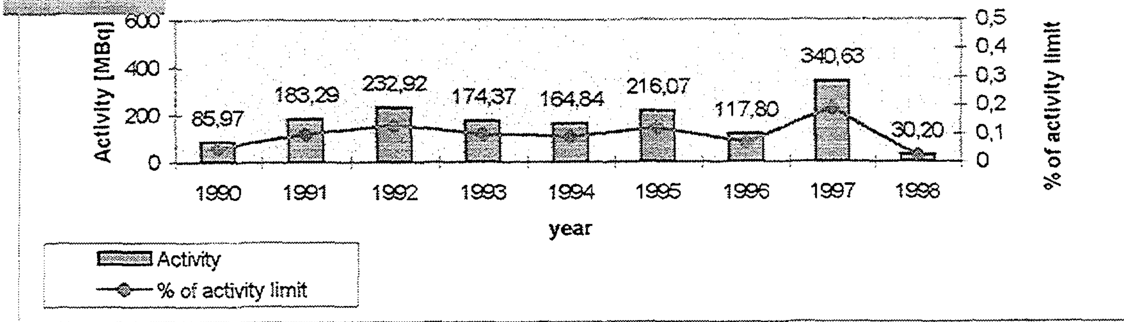


Gaseous discharges V-2

Noble gases



Aerosol



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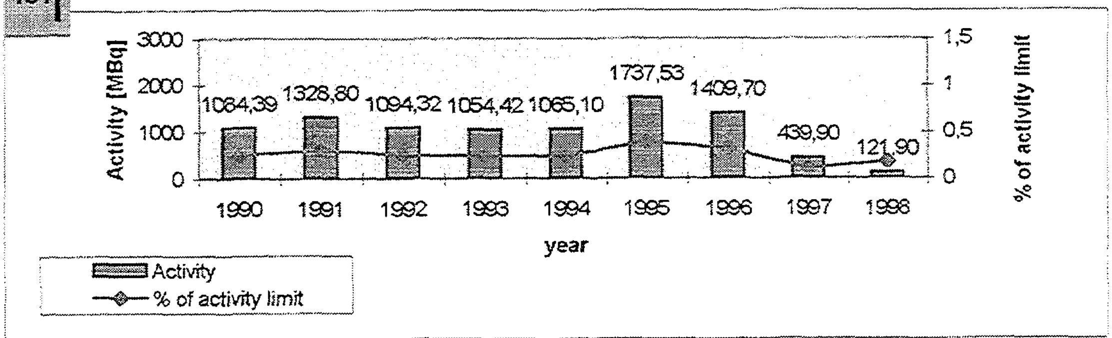

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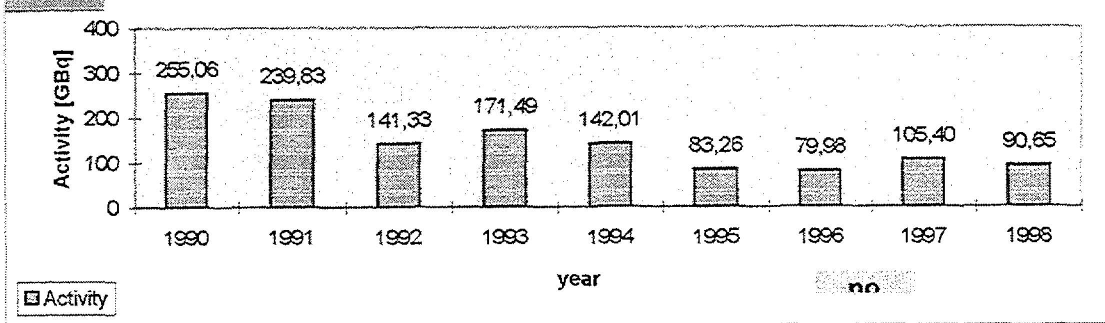
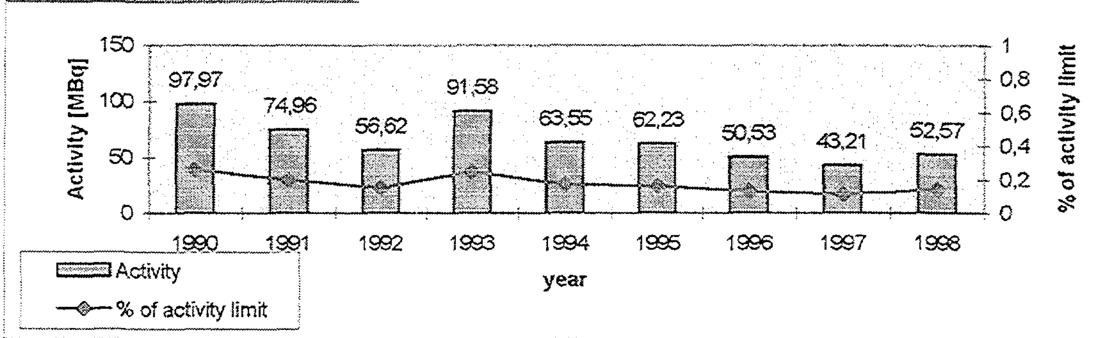
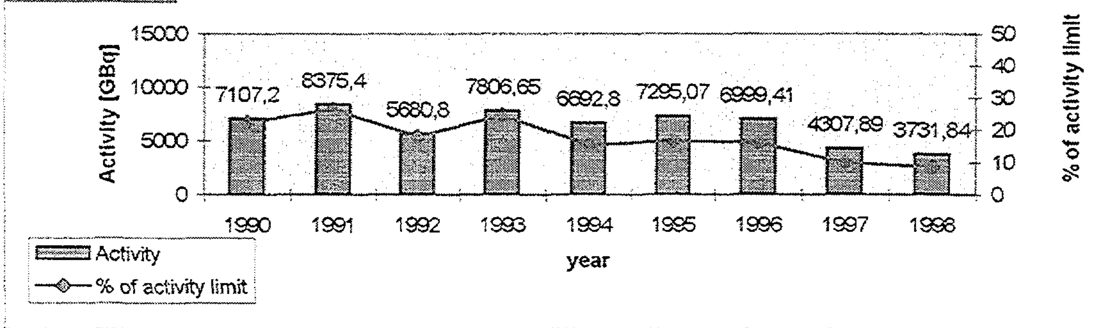


Liquid effluents V-1, V-2

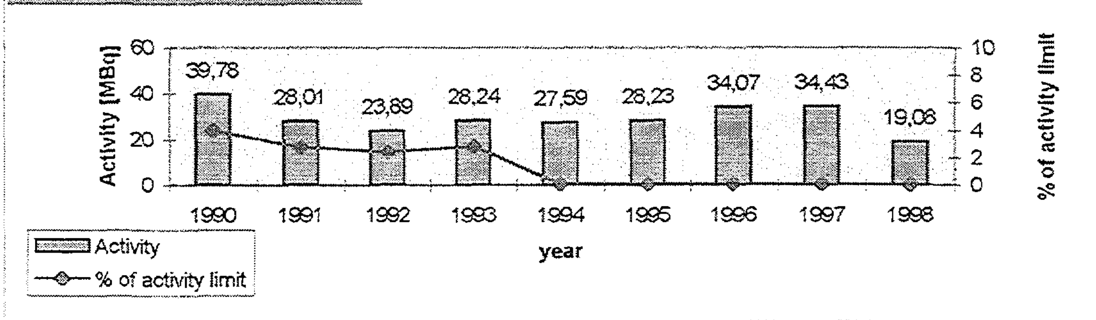
Total activity - V-1



^{3}H - V-1



Total activity - V-2



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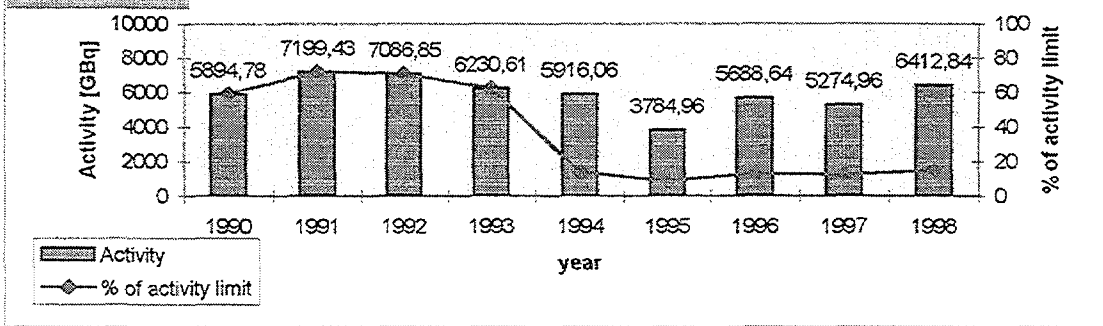
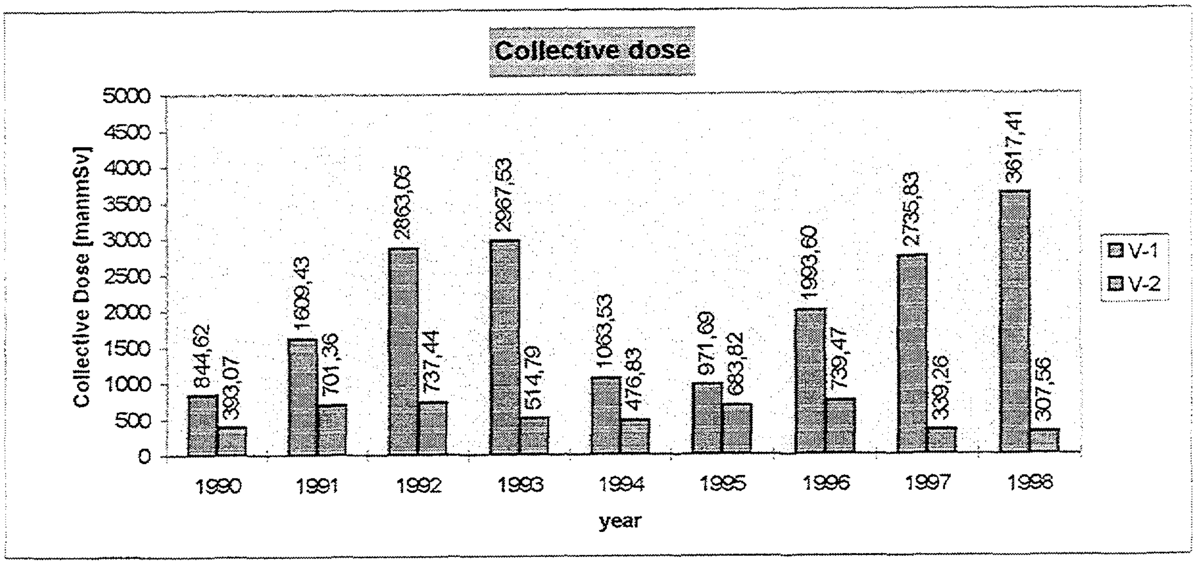
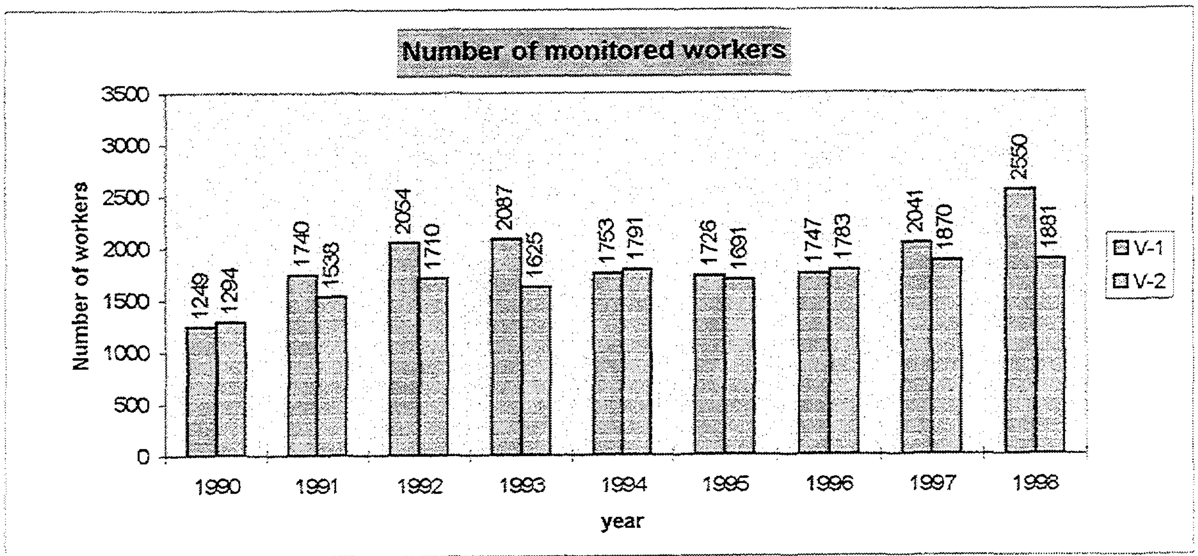
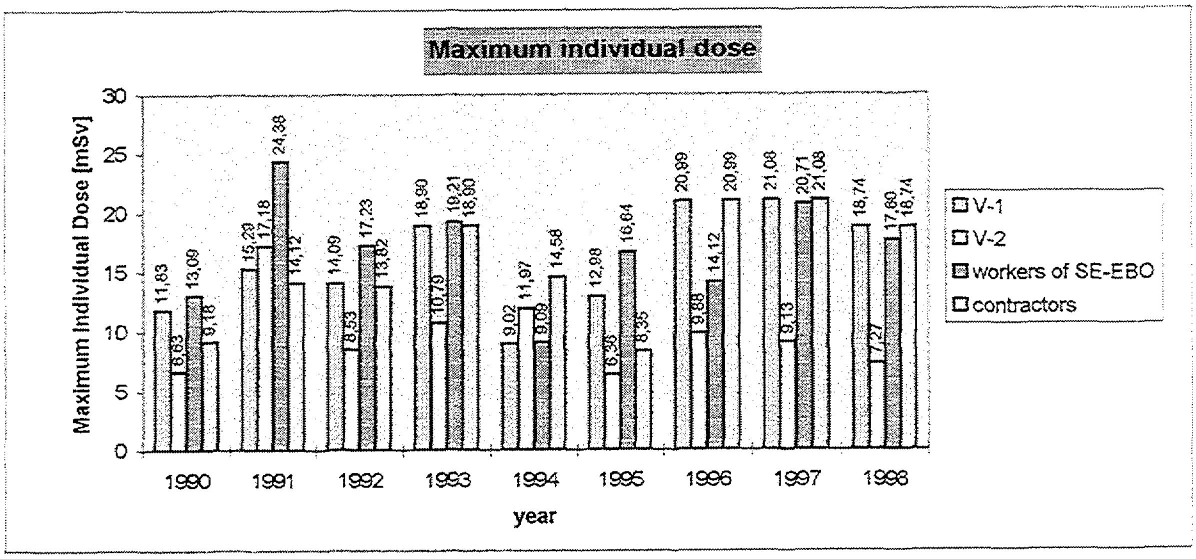


Doses at Bohunice NPP







Dose distribution