Answers to Questions and Comments Raised by Spain

on the

National Report of the Czech Republic

prepared for the purposes of the

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to the
Convention on Nuclear Safety

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SPAIN 1: What is the schedule for the mentioned modernisation programme for Dukovany NPP?. Are all the issues identified by the IAEA 1996 mission included?. Will all the mentioned design improvements be in place before commissioning of Temelín NPP? The generic safety issues for VVER-1000, identified by the IAEA, will or have they been addressed before commissioning?

NPP Dukovany

The Nuclear Power Plant Dukovany is paying prime attention to safety enhancement, i.e. to the „IAEA Safety Issues“ resolution. The first safety enhancements have been included in the so-called „NPP Dukovany backfitting programme“ specified by the Government Resolution No. 309 of 1986. Still before the IAEA-EBP WWER-03 was put in words, a so-called „Minimum List of Measures to Enhance Nuclear Safety of VVER 440/213 Blocks“ had been brought about within the framework of the VVER 440/213 club. The „IAEA Safety Issues“, arisen thereafter, contain a variety of recommendations from this List. The safety aims of the NPP Dukovany voiced today in the upgrading program under the name of „MORAVA“, have been passed through judgement by the IAEA mission on the basis of the IAEA-EBP-WWER-03 document prepared in 1995.

It can be summarised that all of the „Safety Issues IAEA“, categorised II and III, have been included in the upgrading program by the Nuclear Power Plant Dukovany. The "operational" issues stated in the IAEA-EBP-WWER-03 document (13 uncategorised recommendations) is placed under discussions at the NPP Dukovany on separate basis and its level shall be subject to inspection by the repeated OSART mission in 2001. For an outline of the individual safety issues sorted out or being just prepared see the table below.

It is obvious from the graphic information, that the NPP Dukovany intends to gradually resolute altogether 74 identified Safety Issues (categorised I, II, and III).

Until early 1999, 31 measures have been implemented in total, 16 of them in category II and 1 in III.

Of the total number of 40 measures categorised II, 31 will be accomplished by 2002 and as for the 8 findings of IAEA in the category III, all will be realised by the same year. The remaining actions in the category II go hand in hand with implementation of the activity called „Instrumentation and Control System Modernization“. This year in late March, the ČEZ, a.s. made official announcement to the potential contractors to compete for implementation of the Instrumentation and Control System Modernization project.

Schedule of IAEA Safety Issues resolution

Year	Number of Finding	Count per Year	Total	I	II	III
Cont.	AA1 (II), AA2 (I), AA3 (II), AA10 (I)	4	4	2	2	-
by 1998	G3 (II), CI1 (II), CI2 (III), CI4 (II), CI5 (II), CI6 (I), S6 (II), S7 (II), S8 (I), IaC8 (II), EI1 (I), C3 (II), C4 (II), C5 (I), IH1 (II), IH5	24	28	11	11	2

	(I), AA4 ((I), AA7 (II), AA12 (I), AA13 (I), AA14 (I), AA15 (I), IH8 (I), EH3 (III)					
1998	G1 (II), S15 (II), IH3 (II)	3	31	-	2	-
1999	AA5 (I), AA8 (II), G2 (III), EH1 (III)	4	35	1	1	2
2000	S12 (I), EI5 (II), IH4 (II), AA11 (I), S1 (II), S4 (II), S5 (III), S13 (III), C2 (II), IH2 (III)	10	45	2	5	3
2001	S11 (I), S16 (II), RC1(II)	3	48	1	2	-
2002	S3 (II), S9 (II), S10 (II), S17 (II), IaC11 (I), EI2 (I), EI3 (I), EI4 (I), AA6 (II), IH7 (III)	10	58	4	5	1
2003	IaC10 (II), IaC6 (I), IaC7 (II),	3	61	1	2	-
2004	CI3 (II)	1	62	-	1	-
2006	S14 (II)	1	63	-	1	-
2007	S2 (II), IaC1 (II), IaC2 (I), IaC3 (II), IaC4 (II), IaC5 (II), C1 (II)	7	70	1	5	-
2009	IaC9 (II)	1	71	-	1	-
2010	AA9 (I)	1	72	1	-	-
not planned	IH6 (I), EH2 (I),	2	74	2	-	-

NPP Temelín

All of the design improvements issued in the Annex I to the National Report, including the recommendations specified for VVER 1000, will be resolved (overwhelming majority of them has been done already).

All of the Safety Issues up to the level II and III shall be taken into account by the unit start-up and the status of safety issues resolution shall be reviewed before the SÚJB issues its first fuel load-in licence.

Issue Number	Issue Designation	Note
GENERAL		
G1	Component classification	is being addressed
G2	Equipment qualification	is being addressed
G3	Reliability analysis of the safety system class 1 and 2	addressed
REACTOR CORE		
RC1	Prevention inadvertent of boron dilution	addressed
RC2	Control rods insertion reliability /fuel elements deformation	addressed
RC3	Sub-critical status monitoring during reactor shutdown	addressed
COMPONENT INTEGRITY		
CI1	RPV embattlement and its monitoring	addressed
CI2	Non-destructive tests	addressed
CI3	Primary pipeline whipping restrain	is being addressed
CI4	Steam generator collector integrity	addressed
CI5	Stem generator tube integrity	addressed
CI6	Steam and feedwater piping integrity	is being addressed
SYSTEMS		
S01	Primary circuit cold overpressure protection	addressed
S02	Mitigation of a steam generator primary collector break	addressed
S03	Reactor coolant pump seal cooling system	addressed
S04	Pressuriser safety and relief valves qualification for water flow	addressed
S05	ECCS sump screen blocking	addressed
S06	ECCS water storage tank and suction line integrity	addressed
S07	Heat exchanger integrity	addressed
S08	Power operated valves on the ECCS injection lines	addressed
S09	Steam generator safety and relief valves qualification for water flow	addressed
S10	Steam generator safety valve's performance at low pressure	addressed
S11	Stem generator level control valves	addressed
S12	Emergency feedwater makeup procedures	addressed
S13	Cold emergency feedwater supply into steam generators	addressed
S14	Main control room ventilation system	addressed
S15	Hydrogen removal system	addressed

Number of Issue	Issue Designation	Class of Issue
MEASUREMENT AND CONTROL		
I&C01	I&C reliability	addressed
I&C02	Safety system actuation design	addressed
I&C03	Automatic reactor protection for power distribution and DNB	addressed
I&C04	Human engineering of control rods	addressed
I&C05	Control and monitoring of power distributions in load follow mode	addressed
I&C06	Condition monitoring for the mechanical equipment	addressed
I&C07	Primary circuit diagnostic systems	addressed
I&C08	Reactor vessel head leak monitoring system	addressed
I&C09	Accident monitoring instrumentation	addressed
I&C10	Technical support centre	addressed
I&C11	Water chemistry control and monitoring equipment (primary and secondary)	addressed
I&C12	Automatic reactor protection for power and DNB	addressed
I&C13	Power distribution monitoring inside the active zone in load follow operation	addressed
I&C14	Power supply to the plant process computer and I&C systems	addressed
ELECTRIC POWER SUPPLY		
El2	Reliability of diesel-generators	addressed
El3	Protection signals for emergency diesel-generator	addressed
El4	On-site power supply for incident and accident management	addressed
El1	Off-site power via start-up transformers	addressed
El6	Ground faults in DC circuits	addressed
El5	Emergency battery discharge time	addressed
CONTAINMENT		
Cont. 1	Containment by-pass	addressed
INTERNAL RISKS		
IH1	Systematic fire hayard analysis	addressed
IH2	Fire prevention	addressed
IH3	Fire detection and extinguishing	addressed
IH4	Mitigation of fire effects	addressed
IH5	Systematic analysis to floods	addressed
IH6	Flood protection of the emergency electric switchgear	addressed
IH7	Protection against the dynamic effects of main steam and feedwater line.	is being addressed
IH8	Polar crane interlocking	addressed

Issue Number	Issue Designation	Issue Class
EXTERNAL RISKS		
EH 1	Seismic designs	addressed
EH 2	Analysis to plant specific natural external conditions	addressed
EH 3	Man induced external events	addressed
ACCIDENT ANALYSIS		
AA01	Scope and methodology of accident analysis	addressed
AA02	QA of plant data used in accident analysis	addressed
AA03	Computer code and plant model validation	addressed
AA04	Availability of accident analysis results for supporting plant operation	addressed
AA05	Main steam line break analysis	addressed
AA06	Overcooling transients related to pressuriyed thermal shock	addressed
AA07	Steam generator collector rupture analysis	addressed
AA08	Accidents under low power and shutdown (LPS) conditions	addressed
AA09	Severe accident	addressed
AA10	Probabilistic safety assessment (PSA)	addressed
AA11	Boron dilution accidents	addressed
AA12	Spent fuel cask drop accidents	addressed
AA13	Anticipated transients without scram (ATWS)	addressed
AA14	Total loss of electric power	addressed
AA15	Total loss of heat sink	addressed
OPERATION		
OP1	Procedures for normal operation	addressed
OP2	Emergency Operating Procedures	addressed
OP3	Limits and Conditions	addressed
Man1	Need for Safety Culture	addressed
Man2	Experience feedback	addressed
Man3	Quality Assurance Program	addressed
Man4	Data and document management	addressed
PO1	Philosophy on use of procedures	addressed
PO2	Surveilance programme	addressed
PO3	System of communications	addressed
RP1	Radiation protection and monitoring	addressed
Tr1	Training programs	addressed
EP1	Emergency centre	addressed

All of these changes and improvements within the NPP Temelín Project have mainly been induced by:

- Requirements of the State Supervisory Authority (SÚJB),
- Recommendations of the missions, audits, and international feasibility studies (IAEA, NUS Halliburton, ...)
- Component (system) replacements for the reason of their unsatisfactory quality, production cancellation, ...

- Decision of the building/operation organisation (for example technical measures taken as defined within the preliminary stage of the Czech electrification network operation parallel with the one of the countries associated under UCPTE).

In summary, it can be stated that up to now about 90 % of the Safety Issues relevant for Temelín have been fulfilled.

In the Annex to the National Report, there is a reference to the IAEA Mission Final Reports that have put under evaluation the Safety Issues resolution of NPP Temelín. The Mission Report is available via Mr. Šváb, liaison officer of SÚJB.

SPAIN 2: It is stated that SUJB is an independent State administration, but no information is provided on how this is accomplished and preserved. Since SUJB reports to the Government, how it is assured the independence of regulatory policies from Government policies supporting or opposing the use of nuclear energy? Is that reflected in the appointment and removal of the Chairman?

Within the structure of state administration there are four prime attributes of the SÚJB independent position:

1. Even though there is no cabinet member (minister) presiding over SÚJB, the Office has been incorporated in among the Czech Republic central offices of state administration. It means that the Office is independent within the structure of state administration, not being subordinated to any ministry by the Law,
2. Like in the case of any other central of office of state administration of the Czech Republic, the budget of SÚJB is subject to approval by Parliament within the system of state revenue. This means that decision making about the coverage of the SÚJB activities by necessary funds belongs to Parliament - the level of budget, as well as utilisation of the allocated funds must by defended before the Parliament by the SÚJB itself,
3. The SÚJB chairman is appointed and recalled by the government as a body and not by any of the individual ministers,
4. The decisions issued by SÚJB are definite in their nature. This means that no other state administration authority is authorised to intrude in the decision making processes of the Office, and its decisions can be appealed at an independent court of justice only.

SPAIN 3: It is stated that SUJB has its own budget as part of the State budget, and resources are sufficient for fulfilment of the basic functions. Is there enough flexibility to accommodate big licensing efforts derived from Temelín NPP commissioning activities?

Within the Czech Republic, the SÚJB is a central state authority, not being under jurisdiction of any ministry. Like in the case of all other Czech Republic central state authorities, the SÚJB budget is subject to Parliament approval within the state revenue. The decision about financial resources belongs to Parliament - the level of this budget, as well as utilisation of these allocated funds must be defended before the Parliament by the SÚJB itself.

Of course, even the financial resources allocated by the Parliament to SÚJB must respect the confinements given by state revenue. The National Report gives information on the level of them in absolute terms. In general it can be stated that the allocated funds make it possible to fulfil all the responsibilities belonging to the Office by law, thus also those in the field of the questioned supervision over finishing works and start-up of the Nuclear Power Plant Temelín.

SPAIN 4: Regarding the financial responsibilities of the operator for potential damages to the public or the environment. How are they guaranteed? Has the SUJB any review responsibilities before granting the license?

As it is stated in the National Report, the new Atomic Act:

- determines in its head 5 the exclusive and absolute licensee's liability for a nuclear damage caused by operation of this licensee's nuclear installation.
- implements in the Czech Code of Law the Vienna Convention on Civil Liability for Nuclear Damage. The level of the licensee's liability is set forth in the Atomic Act to CZK 6 billions. According to § 33 of the Atomic Act, the licensee is obliged to reach a nuclear damage liability insurance contract, unless the financial coverage for the case of nuclear damage is otherwise available. For details see the Atomic Act attached to the National Report.

Note: In 1998, the Czech Republic has also signed the "Protocol to Amend the 1963 Vienna Convention on Civil Liability for Nuclear Damage", going to implement it into its code of law within 2-3 years.

One of the obligations of the Regulator (SÚJB) is to see to it that the requirements of the Atomic Act are met. It is not however explicitly and by law required of any applicant competing for a license to demonstrate the ability to execute his liabilities should such a nuclear damage arise during the course of licensing procedure.

SPAIN 5: Is their available expertise within CEZ to assess the safety of their facilities independently from the suppliers? Are there other organisations supporting CEZ for this task?

a) Internal situation at the licensee's organisation

The nuclear safety assessment in the nuclear power plants of ČEZ, a.s., provided by the specialised sections within their responsibilities, covers the scope defined in line with the international practice formulated by IAEA. It is a complicated and comprehensive task to determine overall level of a power plant nuclear safety but it can be made easier by braking down this set of problems into several precisely specified elements - safety factors. Every such safety factor is then assessed with use of the usual methods and this means that, in the course of operation, these elements must be constantly reviewed in the form of a series of programs or procedures.

For evaluation of these safety factors, the responsibilities are assigned to the individual company organisational units. The following factors are in question:
- Current status of individual units;
- Safety analyses;
- Equipment qualification;
- Ageing management;
- Safety performance;
- Operational feedback;
- Regulations;
- Organisation and management;
- Human factor;
- Emergency preparedness;
- Impact on surrounding environment.

NPP current status
Assurance of this safety factor means that the current state of the NPP facilities, structures, systems and components being the subject of operation licence and being vital for nuclear safety must be kept and reviewed. Among the most important parts of this safety factor is the verification of the NPP actual state compliance with its design documentation, confirmation of the compliance with the design fundamentals, and verification of the state of modification records (as-built design), verification whether or not there are some changes against initial operating conditions (internal and external) and that they have been taken into account.

Safety Analyses
When the evaluation is carried out, it is necessary to reassess the existing and already elaborated safety analyses with focus on their adequate scope, conditions, and methods and match them to the current state of the NPP, its remaining service life, and current state of knowledge of analytical tools. There is also another set of task connected with this evaluation with the purpose to determine whether the NPP is capable to meet the prescribed safety features and limits and conditions of safety operation (L&C).

Equipment Qualification

Within this safety factor it must be verified that the nuclear safety-vital systems are still qualified enough to perform their design functions as they were specified in the qualification documents at the onset of operation. This verification is being reached by the processes that can establish, document, and retain the proofs of this or that equipment qualification. In practice, a continuous process is the matter, looking into the qualification changes due to ageing, testing, and calibration, modifications, repairs/replacements, wear and tear, or abnormal conditions of operation, etc. The actual qualification-related verification is, apart from others, accomplished in the form of Surveillance Program, operational inspections and maintenance, and it must be therefore checked on occasion of such an evaluation that the inputs from these programs are really used during qualification maintenance.

Ageing Management

For this factor assessment it is necessary to find out whether or not the required safety reserves are kept during ageing, i.e. whether there is or is not an appropriate program of controlled ageing, that it works efficiently enough with regard to maintenance of the power plant future operation. Upon these activities it is necessary to comprehend, predict, and detect the ageing processes, and carry out the corresponding retarding actions. This is why even the controlled ageing program must comprise the related activities, such as Surveillance Program, maintenance, operational inspections, and operational experience feedback.

Safety Performance

The matter is here to review the operational experience, including the safety-relevant events, further to review the records of safety system outages, effectiveness of maintenance, to reassess the tests, checks, replacements/modifications, to review the radiation exposures, radioactive waste output, hazardous conditions and their trends.

For this purpose, the established internal feedback system is employed and evaluated, as well as a part of the Surveillance Program (e.g. a selected information on the state of mechanical barriers, safety systems, etc.). The results of this evaluation must be accumulated up to the end of the power plant life span in the form of so-called safety indicators.

Operational Feedback

The purpose of this safety facto is to check that there are some regulations for operation, maintenance, inspections, tests, and modifications, being of a good standard, that they meet all formal requirements, that they are complete, up-to-date, validated, approved, and available to pertinent personnel. It has to be probed if these regulations are included in the so-called check documentation and are subject to change proceeding, and if any compliance with the safety analyses conditions is ensured.

This mainly pertains to the regulations for:

- normal, abnormal operation and emergency conditions
- maintenance, tests, and inspection (including the regulations for work licence)
- radiation, fire, and mechanical protection, etc.
- further for control procedures for modification of power plant design, modification of regulations and programs.

Organisation and Management

Within this safety factor an overall standard of organisation and control is put under judgement to find out if it is adequate to NS performance. The subjects of evaluation are as follows:

- standard of safety management;
- role of the safety/protection responsibility (in nuclear, mechanical, radiation, fire, technical, and environmental aspects, Safety of Work and Health Protection);
- NPP configuration management;
- Quality Assurance System;
- Technical and contracting support;
- personnel training;
- compliance with requirements of legislation and supervisory authorities.

Human Factors

The goal is here to assess the extent of various human traits and activities and their impact on safe operation of a nuclear power plant, including the trends of related changes. An attention must be paid to this safety factor mainly in the case of operating personnel (shifts), but also, to an appropriate extent by all staff of the power plant.

Emergency Preparedness

In the area of the accident readiness the aim is to find out whether the nuclear power plant itself, but also the local, regional, and state authorities are or are not in possession of adequate tools, i.e. the plans, personnel, sites, and facilities set apart for the case of accident in this NPP and that these tools are properly and regularly tested and exercised and with what results. It has to be established if the safety-related changes in connection with this factor are surveyed and reassesses in the NPP and beyond it, and whether, in respect to these changes the accident readiness is ever updated, including its tools.

Impacts on Surrounding Environment

The subject of evaluation is here, whether or not there is a Surveillance Program for monitoring of plant operation adverse effects on environment and if it is adequately run, and if the relevant remedial actions are executed and with what efficiency. To the largest extent, this pertains to the radioactive nuclide concentration in the gaseous and liquid effluents, adjacent flora and fauna, water, and soil. It has to be verified if the data thus acquired are incessantly compared and confronted with the information from the period before the beginning of the nuclear power plant operation and with permissible limits, and what the trends are like.

b) External technical assistance

For these feasibility studies and analyses, competent independent external organisations are obviously being employed from the Czech Republic and abroad. They are for example the following:

- Nuclear Research Institute, Řež, a.s.;
- Nuclear Power Plant Research Institute, Trnava;
- ENERGOPROJEKT Praha, a.s.;
- Czech Technical University, Prague (ČVUT);
- Brno Technical University, Vítkovice, a.s.;
- Institute for Applied Mechanics, Brno;
- Stevenson and Associates, s.r.o., Plzeň;
- REDYS Prague;
- EDF (France);
- ENCONET;
- NNC (UK);
- ENAC Consortium;
- Data System and Solution (a Rolls Royce and associates and SAIC joint venture).

In some of the specialised quality assurance -related areas where the ČEZ, a.s. has not its own adequately skilled staff or where it is justifiable in economic terms, the selected contractors (e.g. COLENCO, STOLLER, NRI, KPI, and others) can act in lieu of the ČEZ, a.s.

SPAIN 6: Has the NPP safety documentation been updated to meet the new regulations and standards? Are improvements performed by other countries in similar designs being assessed in a systematic way by SUJB or CEZ, to consider the implementation in Dukovany NPP?

The first complete reassessment of nuclear safety (innovated Safety Analysis Report) for the Dukovany units was performed after 10 years of operation using advanced state-of-the-art tools and taking into account operational experience and plant modifications. It was prepared by the utility to fulfil one of the condition of the State Regulatory Body (SUJB) from its decision No. 154 (1991), which established conditions for the 1st unit license for continued operation after 10 years. One of the license conditions requires continually updating ("Living") Operational Safety Analysis Report. During review process of the innovated Safety Analysis Report it was checked if using advanced state-of-the-art tools the safety objectives and all of the legal requirements including the requirements of the Regulatory Body were met. It was also checked if it was accomplished using acceptable analytical methods.

"Living" (periodically updated) Operational Safety Report is now in effect. It documents the state of nuclear safety assurance of the NPP Dukovany units. This report consists of constant unchangeable part (the same for all 4 NPP Dukovany units) as well as of the parts which are updated regularly once a year, always not later than by the end of the next half-year - at the same time for all units. As some parts of the SAR are common for all four units this safety report is based on the complemented "Operational Safety Report for Nuclear Power Plant Dukovany 1st Unit".

A new revision of Operational Safety Analysis Report is now under review process as substantial parts (fuel system design, nuclear design, thermal and hydraulic design, accident analysis) reflects introduction of the new (advanced) Russian fuel.

Upgrading and improvements as well as operating experience performed by other countries in similar designs are being closely observed and assessed by SUJB and CEZ.

As an example we can present impact of malfunctions of RCCAS in VVER-1000 reactors on our licensing process. SUJB concern was aroused by information of control rod malfunctions and measured deformation of fuel assemblies in VVER-1000. Safety concern arises from both the structural deformation and particularly the reliability of control rod insertion. Many cases have been reported since 1992 when the RCCA drop-time has increased above typical value, even above the design limit of 4 seconds and even some rodlets were stuck in an intermediate position in the bottom part of the core The cause has been attributed to friction increase between the control rodlets and the guide channels. The reason for the interference seems to be distortion of the guide channels resulting from higher that design axial loads imposed by the protective tube unit (upper internals) to the fuel assembly caps upon tightening of the vessel head. The control rods insertion reliability of Westinghouse fuel design for Temelín was assessed and reassessed with the conclusion that it should not be susceptible to the control rod insertion problems that have been experienced in some VVER-1000 reactors. But SUJB requested to install technical means, which will be able to detect in time increasing drop-time of control rods.

As another example we can show information on partial clogging on new fuel of Loviisa NPP Unit 2. As the event coincided with introducing improved fuel design with assemblies with Zr1%Nb spacer grids and as also fuel of this type was to be introduced in Dukovany NPP a special evaluation of outlet temperatures of fuel bundles during operation was performed.

Outlet temperatures of assemblies with zirconium spacer grids and that of older type with stainless steel spacer grids in symmetrical position in the core and that of being in the neighbouring position were compared, trends were evaluated and compared. Content of possible reason for crud built-up like corrosion products, colloid and particle impurities were carefully checked. But no coolant flow blockage in reactors of Dukovany NPP was detected.

SPAIN 7: Is there a systematic programme, agreed with the SUJB, for the management of ageing and plant life extension in Dukovany NPP?

In the Dukovany NPP, the ageing monitoring has been so far focused on the power plant key components, which are irreplaceable or hardly replaceable and determine thus the real service life of the whole power plant. How this life is actually consumed in these components. It is reassessed on regular basis and the information about this rate is submitted to SÚJB twice a year. This scheme of actual service life consumption in the case of the power plant key components serve as a basis supporting the onset to the NPP long-time operation.

Currently, the „Life Control Rules“ have been issued, extending the entire program to cover the life control of other mechanic components, further also of the measuring and control components, electric and building components affecting functionally the nuclear safety.

A ten-year revision of the Safety Analysis Report where all the results on component life are included is subject to approval by the SÚJB.

SPAIN 8:   The information on the regulatory activities regarding radiation protection is very limited. Details on the surveillance and control programmes being performed by SUJB to ensure the adequacy of NPP practices are needed.

Inspections on radiation protection of NPP are performed by a specialised group of SÚJB radiation protection inspectors. These inspections are performed systematically according semi-annual plan of inspections: The results of inspections are evaluated not only by the inspectors themselves, but also by the top management of SÚJB on a regular semi-annual basis. Both measurements of radiation doses to workers and environmental measurements of the NPP are supervised by SÚJB and, in addition, a national-wide independent radiation monitoring network is operated under the co-ordination of SÚJB.

SPAIN 9: Is there a legal document assigning the different responsibilities to the agencies involved in off-site emergency response?. If this legal document is the Nuclear Act No. 18/1997, has the Dukovany emergency plan been updated? Since the Act is very recent, has the new organisation been tested by drills?. If so, what were the main findings.

Ad a) Major responsibilities of the authorities included in the system of off-site emergency preparedness, Ministry of Defence, Ministry of Interior, Ministry of Health, District Offices, and SÚJB are determined by the Atomic Act. Further responsibilities are obvious from the constitutional law on security CR, law on defence, law on fire protection, law on the Police, law on District Offices, and from some of the governmental and administrative acts. Further clarification of responsibilities on the side of individual persons, corporate bodies, communities, District Offices, state administration central authorities, ministries, and government will form a subject of newly proposed act on crisis procedures and integrated rescue system (Crisis Act) assumed to attaining its force on Jan. 1, 2000.

Ad b) The NPP Dukovany has already updated the on-site emergency plan (EP) with respect to the requirements established by the Atomic Act No. 18/1997 Coll. and SÚJB Decree No. 219/1997 Coll. (on particulars for assurance of emergency preparedness of nuclear installations and workplaces with ionising radiation sources and on requirements for contents of on-site emergency plan and the emergency code). This on-site emergency plan was approved by decision of SÚJB No. 228/1998 of May 25, 1998 and entered into force on July 1, 1998.

On the basis of the requirements consistent with the regional project of IAEA RER/9/050 “Harmonisation of Regional Nuclear Emergency Preparedness” and the additional requirements of the local authorities were updated in the same parts of the on-site emergency plan of the NPP, Dukovany. There were performed changes concerning the forms of how to announce a nuclear or radiological accident (initial and additional information) determined for the SÚJB and local authorities. These changes of the on-site emergency plan were approved by the SÚJB decision No. 84/99 of January 27, 1999 and entered into force on February 15, 1999.

Ad c) Prior to this above decision day of effectiveness, the Dukovany NPP has conclusively acquainted all its staff members with its on-site emergency plan, having also trained its appointed staff members in other activities in the case of potential emergencies. These matters have been verified by the SÚJB inspections. In connection with the training sessions carried out, the frequency of training courses in the practical procedures of the shift emergency staff has been increased. In contrast to only 6 exercises in the first half of 1998, the Shift Emergency Staff went through 18 of them by the end of the same year, training also co-operation with external organisations.

SPAIN 10: The SUJB has different responsibilities in case of radiation emergencies and "provides" for the activities of the Emergency Response Centre. How is this centre operated and what means are available?. Is SUJB trained periodically in the operation of the centre?

Ad a) Emergency Response Centre (ERC) came to life at the SÚJB on the basis of the Government Commission for Radiation Accidents requirement to provide for its technical support. Several rooms have been set apart for this purpose, interconnected by a separate computed network isolated from other sections of the Office and connected via the central data archive to the source of information. Basic serviceability of ERC falls under responsibility of the „Emergency Preparedness“ Department reporting directly to the SÚJB chairman. The activities at the ERC working centre are co-ordinated by the SÚJB Crisis Staff, consisting of the management group, nuclear installation status assessment group, group set apart to estimate the radiological consequences of emergencies, and group of logistics.

Ad b) At present the ERC has at its disposal the following software tools:

- KBF monitoring system (Critical Safety Function programs) which covers systems and parameters of NPP Dukovany vital for safety;
- RTARC (Real Time Accident Release Consequence) program developed by the VUJE institute calculates the actual and potential consequences in terms of the dose rates received by the nearby population, on the basis of the previously identified source condition and actual weather conditions;
- SESAME-VVER programs tools for first-approach quick assessment of consequences at the beginning of the accident and for determination of the source term;
- METEO programs tools based on the on-line data from Hydro-meteorological Network for assessment of weather conditions;
- SVZ program tools (system of timely finding) assessment of radiation situation in Czech territory.

Ad c) The SÚJB is paying a continuous attention to this ERC activities and related problems. This is partially connected with completion of the construction works on the ERC and providing it with necessary equipment, further with the operating procedures preparation for each member of the emergency management, nuclear safety evaluation group, radiological consequences evaluation group and logistic group. Part of procedures was already elaborated, part will be obtained by means of IAEA for implementation into ERC, and another part is under preparation by National Radiation Protection Institute. The training on ERC is also adapted to this situation.

SPAIN 11: A meteorological station at 3 Km. from Temelin is mentioned in the report. Is there not a station in the NPP site for local measurements?

The meteorological station at Temelín (3 km away of the NPP Temelín site) was built here to meet solely the needs of the NPP Temelín in the field of measurements in this locality. With its outfit and credibility of achieved results in mind, this station is meeting all requirements of the up-to-date meteorology and its activity is further supported by the representative results against the inhabitants in the NPP neighbourhood. Thus, there is no meteorological station just within the site.

SPAIN 12: No mention of the approach towards radiological consequences of the accidents is made in this article. Is this aspect addressed in the licensing process?

Protection against the accidental release of radioactive materials into environment and preparedness to mitigate the radiological consequences of a potential accident are essential characteristics of all Czech nuclear-related laws. Such a concept is a generic part of the definitions of crucial terms set down in current Czech Atomic Act, namely “nuclear safety” and “emergency preparedness”. Nuclear safety means the condition and ability of a nuclear installation and its servicing personnel to prevent the uncontrolled development of a fission chain reaction or an inadmissible release of radioactive substances or ionising radiation into the environment, and to reduce the consequences of such accidents. Emergency preparedness means an ability to recognise the onset of a radiation accident and, upon its occurrence, to carry out measures specified in the emergency plans.

The same terms, with the same meaning, are used in the National Report, unfortunately without explanation and therefore little bit unclear for foreign readers. In such context, any text of the National Report mentioning “requirements for nuclear safety” and “requirements for emergency preparedness” include implicitly protection against the accidental release of radioactive materials and preparation to mitigate the radiological consequences of an accident.

These aspects are addressed in all stages of licensing process: from siting criteria through design and construction to operation of nuclear installations. Details are set down in appropriate operational Limits and Conditions, Monitoring Plans and Accident Plans approved by SÚJB within licensing. Similar approach was applied also according to previous legislation valid in the past at the time of siting and designing the first Czech NPP Dukovany.

SPAIN 13: Only LOCAs are mentioned when referring to design basis accidents, please specify what other accidents are considered and reviewed as DBAs. Is there some generic policy by SUJB for addressing severe accident response by current plants?

NPP Dukovany

The DBA List for the **Dukovany NPP** is similar to that used by any western NPPs. The following groups of PIE are what matters here:

- change in radioactivity;
- defective coolant flow rate;
- defective heat extraction in the secondary circuit;
- loss of coolant in primary circuit (LOCA including SGTR, SGCR);
- broken steam pipeline (including the breakage on the main steam collector);
- broken feedwater pipeline;
- defective electricity supply;
- faults in fuel handling;
- faults in auxiliary systems;
- faults in the radioactive waste management system;
- internal flooding;
- internal fires;
- external hazards;

There is, however, a peculiarity in the VVER 440/213 NPP design: within the limits of PIE a so-called „maximum design accident" has been picked and defined, resting in a guillotine-like cracking in the primary circuit (Js 500) in the inseparable part at the reactor input (see chapter 13.1.2).

The safety analyses in the OSAR have been prepared for the DBA scope. The severe accidents have been analysed within the limits of other research projects. Considerable number of these analyses was carried out within the limits of the state-funded project of „NPP Safety" finished in 1992.
The project of PHARE 4.2.7a „Beyond DBA Analysis" has been focused on severe accidents and the methods how they can be sorted out, finished in 1998.
Within the grant assigned by the US government the PSA-2 Study has been prepared in limited cope, being an analogy to the IPP (Internal Plant Evaluation) worked out for the American NPPs. This Study has been prepared by the American company of SAIC in co-operation with NRI Řež as its subcontractor. The Study has been finished in the last year (see chapter 9.1.2 of the National Report).

In the last year this limited Study of PSA-2 has been updated to be based on the existing current Study of PSA-1. The work on SAMG and proposed preventive and mitigating measures will be based on the conclusions of the PHARE 4.2.7a project.

NPP Temelín

For Temelín NPP a revision of deterministic safety analysis was performed taking into account design changes. These analyses were performed with use of US methodology and criteria. Safety analyses documented in Chapter 15 of the revised Preliminary PSAR (Amendment) conforms to those required by US NRC RG 1.70.

SÚJB approach to Severe Accidents

Initial step was taken at the turn of 80's and 90's by the former ČSKAE (predecessor of SÚJB) which, within the limits of the state-funded project of „NPP Safety“ finished in 1992, has ordered a considerable number of the accident and emergency analyses. At the same time, the SÚJB has provided its support to the PHARE 4.2.7a project called „Beyond DBA Analysis" finished in 1998 and focused at the same time on the severe accidents.

Further the SÚJB has suggested and agreed with the licensee to prepare the "severe accident management guidelines" (SAMG). The work is actually a continuation of the "Emergency Operating Procedures" (EOP) that have already been prepared as a condition given to the operating agent by the Supervisory Authority. The choice of heavy accidents subject to evaluation observed the results of the PSA Study, as well as the top-priority preferable criteria. The Temelín NPP is already working on the SAMG, the Dukovany NPP is about to embark on the work in this year as soon as the work on EOP is done. The results will be used in the system of emergency planning.

SPAIN 14: Are the design improvements planned for Temelín based on a proven technology, considering the interface between two different technologies? Have there been planned special provisions to test this interface?

Improvement in the design and the interface between them are under control at the stage of designing already when the necessary interfaces and co-ordination is provided by the Architect Designer organisation for the Temelín NPP - Energoprojekt Praha. Another inspection take place within the framework of the works on the final designs, and then during the course of starting and adjusting works (according to a special programs) where these interlinks are subject to verification on material basis.

The technologies and equipment replaced on the basis of changed design documentation of the Temelín NPP have already been used in other branches of industry or they are in service at other nuclear or conventional power plants or they have been tested on trial stands under really existing conditions.

Concerning the substantial changes in the Temelín NPP design documentation (like those of I&C in the WELCO contract), the interconnections with the existing technologies are moreover established in the course of factory tests already, and in the case of some selected safety systems moreover, the correct function is being verified within the programs of software and main control rooms independent validation and verification.

SPAIN 15: What is the granted license time after the commissioning period is completed? Is there a specified procedure for license renovation after every outage? How are long term safety issues addressed in a one-cycle license approach?

Ad a) Under the Act No. 50/1976 Coll., (Construction Act) and in accordance with Act of 24th January 1997 of Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Act) a license granted by the Office is required for location of nuclear installation, construction of nuclear installation, but also for particular stages of nuclear installation commissioning defined in implementing regulation, operation of nuclear installation. That means that besides license for siting, construction and operation, a license is needed also for a number of other activities for individual stages of nuclear installation commissioning (as well as for reconstruction or other changes affecting nuclear safety). In such a way the licences for individual activities have to be granted.

No time restriction for operation license (so it is permanent in the sense of the way of utilisation) is given by the law but the individual activities connected with operation require further licenses/approvals for these activities. Although the operation license under the Act is not time-limited, during operation the Regulatory Body (SÚJB) issues licenses for restart of a nuclear reactor to criticality following each nuclear fuel reload, based on review of the documentation submitted in accordance with Appendix E to the Atomic Act.

Ad b) A license granted by the Office is required for the restart of a nuclear reactor back to the critical state after refueling in accordance with Act of 24th January 1997 of Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Law) and about Alterations and Amendments of Some Legislation (see Chapter III §9 (1) e) Licenses for Individual Activities)

As a part of license granting procedure for a new reload previous fuel cycle performance is being evaluated (before and during the outage) including comparison of predicted and actual parameters, measured activity of primary circuit, etc. The parameters of the new reload pattern of the proposed new cycle are being checked, especially from the point of view safety of analysis and meeting the safety requirements (fuel design limits). Start-up test program, reload program, functional ability tests are reviewed. Periodic safety reports, which are used for input into the NPP Ageing Management Program, submitted to the regulatory body are also reviewed. Overall inspection (and some special directed inspection) of preparedness of the plant facilities and personnel for nuclear fuel reloading is performed.

SPAIN 16: Which is the SUJB review and control process for design modifications being implemented by the operator? Is there a permit before completion of the modification? Are their criteria to exempt the operator from this procedure?

In line with the operator's applicable program procedures, the safety impacts of every proposed modification are placed under independent judgement by the technical development specialists, staff members of the Nuclear Safety Department, at first. Afterwards they are submitted to the locally assigned SÚJB inspectors for their assessment.

All safety-relevant modifications are considered in the „Technical Safety Commission".

The SÚJB approval is applicable to the facilities being in the List of Selected Items.

This List has been put together by the architect/design engineer with regard to the criteria assigning to this or that facility a level of seriousness in the field of nuclear and radiation safety as according to the current Czech legislation. Then, in line with the requirements under § 9 of the Law 18/97 Coll. „on Nuclear Power Peaceful Utilisation..." it is necessary for actual implementation to obtain the SÚJB licence. Only with the licence in hand, the modification can be implemented.

The SÚJB licence on the basis of which a modification can be implemented is subject to submission of the safety documentation provided that the SÚJB has then taken its affirmative attitude to it. In this documentation the Quality Assurance Plan must always be included where the quality assurance requirements are rooted with a schedule how they will be met from preliminary stage up to the implementation itself, including the proofs saying that the proposed modification will not have any adverse impacts on the nuclear power plant safety. For particular scope of this safety documentation see Appendix F to the Law 18/97 Coll. and related Executive SÚJB Regulation No. 214/97 Coll.

Otherwise, where the modification has nothing to do with the List of Selected Facilities, the preparatory phase to the modification and its implementation at the operator's are subject to widely applicable quality assurance principles. In the domain of modifications, these mainly rest in two-level review of all the document arisen in the course of the modification preliminary phase and in the quality audits performed on regular basis at the contractors figured out by the ČEZ, a.s. company.