Answers to Questions and Comments Raised by France

on the

National Report of the Czech Republic

prepared for the purposes of the

First Review Meeting of Contracting Parties
to the
Convention on Nuclear Safety

Vienna
12-23 April 1999

FRANCE 1: The report indicates that many safety evaluations of Dukovany and Temelín NPPs have been performed using deterministic and probabilistic approaches as recommended by INSAG 8. But the report does not give the results of these evaluations, that is to say, the importance of the weaknesses to maintain an adequate defense-in-depth. In particular the report only gives the core melt frequency, which is not the most important result of a PSA, the role of which being to find the design and operating weaknesses of the plant.

FRANCE 2: More information would be appreciated in order to establish a link between the results of the safety evaluations and the list of the decided modifications which have to represent all practicable improvements as stated in Article 6 of the Convention.

FRANCE 3: More information should also be given on the difference in the safety requirements of Dukovany NPP and Temelín NPP which should be in conformity with the ones of modern plants.

FRANCE 4: Could the Czech Republic provide such additional information ?

What concerns the NPP Dukovany, safety measures included in the "MORAVA" modernization program have been proposed in line with extensive analyses and judgements carried out in the period from 1992 to 1997 both with use of internal resources and those of international missions of evaluations. The following can be included in among the main evaluating actions:

1.	Internal and external audit and the IAEA mission thereafter to process the conclusions of the inter-budget IAEA program of Safety Issues (IAEA-EBP-WWER-03):	■ 40 measures categorized II (in line with the IAEA classification). 8 measures categorized III and II (in line with the IAEA).
2.	Analyses carried out within the work of operational regulations for accident elimination	■ 17 additional modification proposed (other proposed safety measures are included under 1)
3.	Analyses carried out within the PSA 1,2 (probabilistic safety analyses):	■ 5 additional modification proposed (other proposed safety measures are included under 1 and 2)
4.	Evaluation of the PPDU safety after 10 years of operation (so-called Operational Safety Analysis Report):	■ proposed safety are included under the points 1 through3)
5.	Requirements of State Supervision over Nuclear Safety:	■ for required safety measures see 1 through 4 where they are included

Enumeration of all safety evaluation results and of the related modifications would bee all too extensive, though some of the major conclusions are listed in the below table:

Safety evaluation results:	Proposed or executed measures:
In some of the LOCA emergency modes, the sumps in the SG box might get clogged with the torn down insulation.	Modification of the sieves at entry point into the SG box sumps so that they could not be clogged.
The pipelines of the main emergency and super-emergency feedwater into the SG is in the shared area of the MCR room +14,7m (PoE +14,7m).	The super-emergency SG feedwater pipeline must be relayed somewhere else
Some of the electric drives of steam and water line valves are not qualified for the environmental effects in the case of HELB.	Replacement of water and steam valve electric drives by those qualified for PoE +14,7m.
Inadequate protection against the effects in the case of cracked water/steam line at the PoE +14,7m.	Anti-whipping protection must be installed as well as the barriers against jetting water.
In the case of some LOCA interfacing accidents the coolant will leak into the room of the water circulation pump electric motors and cannot get back into the SG box.	Installation of the drainage route from the room of pump driving motors to the SG box.
In some cases there is an inadequate signal diversity of the reactor protections	The new protective signals have been proposed, derived from the high-pressure in PO, from high temperature at the reactor output, and from high level in KO.

At the time of engineering phase and construction of the Dukovany NPP, the safety requirements came out of the Czech and Russian regulations applicable at that time. The current safety requirements are built on the legislation stated in the addendum to the National Report. The Regulatory Body (SÚJB) is matching the level of nuclear safety against these current requirements.

Temelín NPP underwent in the end of 80's and beginning of 90's a series of expert appraisals and assessments:

- SÚJB review of Preliminary SAR;
- Expert reviews organized by the IAEA;
- Audits organized by licensee (NUS Halliburton).

Criteria for these assessments were those included in Czech legislation (see the National Report), IAEA guides, and engineering standards usual for western power plants at the end of nineties.

Results of these independent appraisals finally led to considerable set design improvements. List of 60 most important design changes is a part of Annex one to National Report. Revised deterministic analyses were than performed to reflect these design changes. Analyses were documented in the Amendment to the Temelín NPP Preliminary SAR. Majority of the deterministic safety analyses was performed by Westinghouse (supplier of new core design and I&C systems) using the US standards and methodology and IAEA recommendations. Results of this new safety analyses demonstrated compliance of NPP Temelín design with criteria set both by US and Czech legislation. PSAR Amendment was submitted for SÚJB review, which is now being completed.

Utility is now completing revision of PSA Level-1 to demonstrate the appropriateness of implemented design changes.

FRANCE 5: How does the Czech Republic ensure that each licensee has an adequate strategy and action plan in place to deal with the year 2000 safety issue?

In matters of the Y2K a Government Resolution No. 108 of Feb. 1, 1999 has been issued within the Czech Republic, where an urgent need to sort out the Y2K issue was stated and in which the Office for State Information System was appointed to act as a National Co-ordinator.

With respect to the nuclear safety, the State Office for Nuclear Safety has carried out inspections at the licensees, in particular in the Nuclear Power Plant Dukovany (under operation) and in the Nuclear Power Plant Temelín (under construction).

The operating organization, i.e. the company of ČEZ, a.s., has paid attention to the Y2K problem since 1997. The problem is disentangled within the entire organization. The assessment has run under the corporate team and in parallel by the individual plants.

For the Y2K problem resolution, the manuals have been prepared in writing and special funds allocated.

Readiness of the equipment deemed important from the nuclear safety point of view shall be documented to the regulatory body once the remaining analyses, tests, and their evaluation are available, i.e. in June 1999. At the same time, the plan of emergency measures shall be submitted for the case of any unexpected and unpredictable events.

FRANCE 6: Legal regulations supplemented by the Safety Guide published by the Regulatory Body cover the area of professional training of personnel. Training is carried out in the training center of Dukovany. Simulator training is carried out in the training center at VUJE (Trnava, Slovakia). When will the Dukovany simulator be in operation?

Currently, the ČEZ, a.s. is training its operative management personnel of NPP Dukovany on the VVER 440 full-scope simulator with use of the VÚJE Trnava Training Centre. This training has a form of a basic course (part of the basic preparation of the new employees before the authorization of SÚJB is awarded to them), periodic course (for the staff members active as operative managers), and re-qualification course (being a part of education on transition between the individual posts within the operative management staff). The obligation to take this training is rooted in the applicable Czech legislation (SÚJB Regulation 146/1997 Coll.).

The preliminary works prior to onset of the training on the multifunctional simulator developed for NPP Dukovany within the limits of the PHARE programme have now reached their culminating point. This preparation comprises fulfillment of all legislative conditions necessary for accomplishment of the specialized training of selected staff (see the Law 18/1997 Coll. and SÚJB Regulation 146/1997 Coll.), including elaboration of the necessary training materials and training documentation (training programs, task scenarios, etc.). Once the training is begun on the VVER 440 multifunctional simulator (in the 2nd half of 1999), a part of the basic training (within the scope of 2 weeks) shall be transferred to this VVER 440 machine.

Construction of the VVER 440 full-scope simulator of the type „main control room replica“ is assumed to finish in 4th quarter of 2000 when it ought to be taken over by the NPP Dukovany. Just after site acceptance tests, the trial operation will follow connected with simulator testing and practical training of instructors scheduled to make training on the simulator. With the legislative requirements being fulfilled (training programs, scenarios of tasks, system of rating), the training itself can be assumed to start on the full-scope simulator in the 3rd quarter of 2001. The basic periodic and re-qualification course shall then be accomplished to its full extent on these VVER 440 full-scope simulators.

FRANCE 7: Could the Czech Republic provide more information on the man-machine interface (indicators, aids for operation, automatism, safety panel, ...) ?

At the main control room (MCR) of the NPP Dukovany the annunciating system is based on easy-to-view mimic panels displaying the signals varying in function and priority in different colors accordingly (the emergencies, for example, are displayed in red). The color indications appear along with beeping (which, respecting the priority of the signal, might be of different types). Due to this concept, it is easy for the MCR operative personnel to adapt themselves quickly and get their bearings. Besides this indication, the data are expeditiously listed out on the printer, following the chronological sequence of occurrence.

At the MCR, there have been and still are some moves in some of the indicators, meters, and control as the MCR personnel consider the new positions better themselves and enhance ergonomic design. The obsolete secondary meters are being replaced by the new and more advanced ones (like plotters combined with digital displays, etc.).

There at the MCR, an „Operator Support System" (OSS) has been installed, representing vital assistance to the MCR operators in action. With use of this system, for example the following can be accomplished:

- Full parameter archive can be retrieved;
- Whatever curves can be compiled complete with the parameters plotted against time;
- Various calculations (critical states, shutdown concentration of H_3 BO_3 and others);
- Watching for various trends (like cool down trends or those of PO reheating);
- Immediate access to the digitized operational documentation;
- Office and paperwork;
- Others.

The in-core measurement display system is being step-by-step replaced by the advanced system SCORPIO, developed with the limits of „Halden Project". It makes the operator's choice wider, providing him more information in a very clear form. At present, the MCR Safety Panel is under development. The most vital readings will be concentrated on the panel, making it possible to interpret comprehensively all of the critical safety functions.

To the MCR, an all-inclusive fire panel has been added, with its interpretation and control unit. Any fire can be promptly localized on this panel.

Also a so-called „Isle Mode" software package have been implemented and tested at the MCR of NPP Dukovany, including all associated automatic systems.

The NPP, Temelín shall have the system similar to that at the Sizewell B NPP.

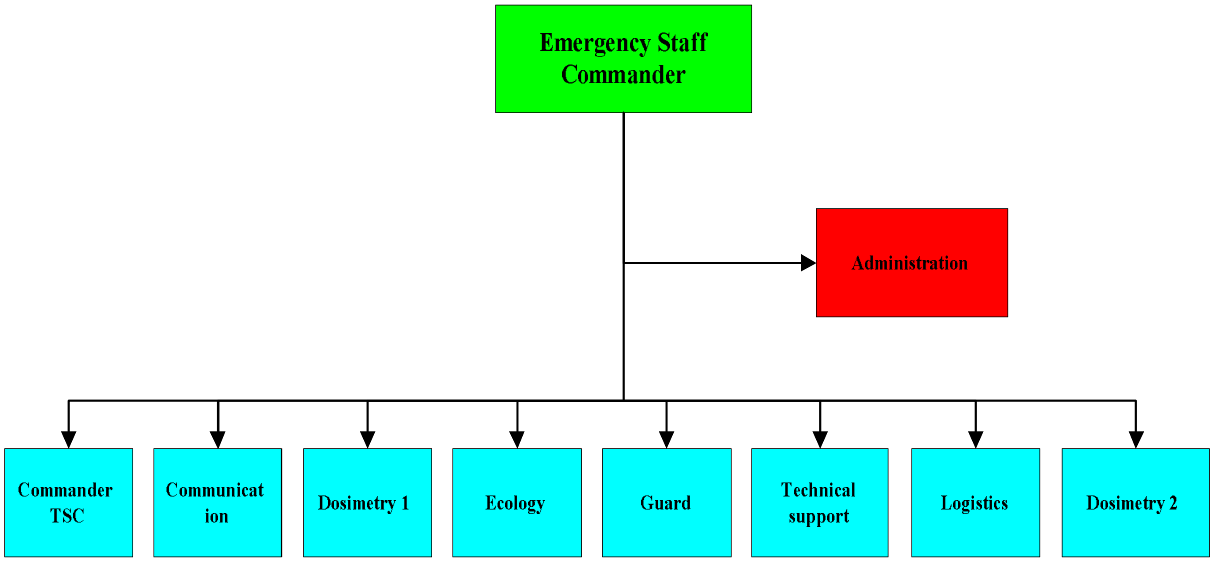
FRANCE 8: Supplementary information would also be appreciated concerning the organization for normal and emergency operation.

Normal operation of each unit is controlled and co-ordinated from its MCR. Each unit, however, has also its emergency control room (ECR), separate from and independent on the MCR. From this ECR, the unit can be shut down and cooled down to its safe status. This shutdown and cool-down procedure is exercised on regular basis and carried out actually when the unit is shut down for fuel replacement.

Transfer from MCR to ECR is prescribed when the former is either endangered or inaccessible to operators, like because of a fire - for such a case the MCR is, beyond the limits of original design, armed with the Halon extinguishing system and respirators for the MCR personnel.

In case of an event within the relevant category according to the on-site emergency plan, the shift engineer will call up the shift emergency staff (non-stop available), warn the personnel in an appropriate way and the external authorities.

It is within responsibility of the emergency staff to mobilize other groups and to provide necessary support to the shift personnel exerting efforts to settle the situation, and provide for communication. For the NPP, Dukovany chart of organization sees the below figure.



For details see the National Report.

According to the announcements made by the SÚJB, the on-line communication of selected data about the status of the NNP's Dukovany units to the SÚJB is now being activated. On the side of SÚJB, the data are interpreted/assessed independently. Once the trial operation of this communication link is over, the normal operation will follow within which the data transmission will be activated for regular verification once a week.

An analogous scheme and system interface with external authorities will also be implemented for the NPP Temelín.

FRANCE 9: Shutdown situations have particular features concerning human factors : are there specific measures (procedures) relating to shutdown situations ?

Yes, there are specific operational procedures available at the Dukovany plant for shut down conditions. These procedures have to be followed by operations personnel.

In addition, shut down and low-power PSA is just now being completed for Dukovany NPP. Results of this analysis will be used for the future safety enhancements.

FRANCE 10, report ref. 9.1.2, on systematic monitoring and periodic assessment of nuclear safety of nuclear power installations: The report states : "The work performed during units outage is checked by the managers of other centers whose personnel or whose subcontracted personnel has been involved in such work". Does "center" mean "reactor unit" ? If so, is such a practice a QA requirement?

A „center" in this context means an organizational unit, such as e.g. a department, but not reactor unit.

FRANCE 11: ***It also mentions: "Since 1995, the unavailability is monitored at all four Dukovany units. Impact of individual components unavailability on nuclear safety is assessed using absolute value of the core meltdown probability and cumulated risk value which is a product of the increase of the core meltdown probability above the basis level and duration of the component unavailability". ... "Results of such assessments-trends of the indicators for 1997 are shown in appendix 3 to this National report". These trends are given in annex 3 for the Dukovany NPP. Could the Czech Republic give separate data for each unit, in order to allow a more precise evaluation of these units?***

FRANCE 12: ***The same remark also applies to SUJB 1997 Annual Report which gives in annex 3 an assessment of the Dukovany NPP using the modified SALP system.***

The Dukovany NPP is being continuously assessed on the basis of a set of safety indicators prepared by SÚJB under participation of NPP Dukovany. This set illustrates a picture of the Dukovany NPP general standard, reliability, and safety as a whole as there is a variety of indicators that cannot express any unit fire (number of fires at NPP, of the liquid discharge, gaseous discharge, collective dose equivalent, etc.). Due to that all of the NPP Dukovany units are identical (in technologies, safety systems, auxiliary systems, technical standard,...), this reassessment will include a prolonged period of operation and these indicators will give a better account of the entire power plant condition. Also a set of indicators is used in various areas for the individual units. The deviations of the results between the individual units and their personnel are statistical in nature and, when interpreting them, we came across fact that would result, for example in the last year, in any remedial action for this given unit only.

Different is the situation in the domain of the events where the event are specific just for a given type of equipment and a particular unit. Therefore we do our best to accomplish the remedial measures for all four units (the same is the EDF approach for all of its run power plants).

FRANCE 13 on Deterministic nuclear safety assessment (Operational Safety Analysis Report): The report states: "In 1991, the state Regulatory Body (CSKAE) in its decision n°154 established conditions to be fulfilled by the Dukovany Operator to obtain a license for the 1st unit continued operation after 10 years. One of these conditions imposed an obligation to submit an innovative Safety Analysis Report which should present evidence of the unit safety proved by most advanced, state-of-the art tools and at the same time considering the existing operational experience". Does this mean those differences between older design criteria and current standards had to be taken into account (e.g. differences between older and current seismic design criteria)?

The revision of the Safety Analyses Report after 10 years of operation was prepared by using state of the art tools (e.g. new calculation codes). Original design criteria and current requirements derived from valid Czech legislation and internationally accepted standards were used for this assessment (for current valid legislation see appendix to the National Report). Results of these assessments confirm that the original design criteria and current requirements are basically met. In case of any deviations, SUJB required to resolve this situation in order to achieve required level of safety.

This new Safety Analysis Report has been evaluated by SUJB which has, on the basis of it, issued its Resolution No. 197/95 where the SUJB has ordered to the operator in these 97 conditions to resolve the discovered drawbacks and to set the terms of time for their final disposition. In response to this SÚJB Resolution the operator then prepared the MORAVA program of equipment modernization, upgrading, and increase in the PPDE safety.

As an example, a questioned „issue" of seismic criteria can be stated. In the context with criteria of the SÚJB Resolution No. 197/95, the operator must re-qualify its equipment with regard to the NPP Dukovany siting for the new seismic resistance values under SS 50 - SG - S1.

FRANCE 14: ***It also indicates : "On the basis of the Operational Safety Analysis Report, the SUJB by its decision n°197 in August 1995 has issued 2 year license for continued operation of Dukovany unit 1. This consent was conditional on the fulfillment of 97 requirements. At present, based on review how these obligations are being kept, the SUJB issues its license for continuous operation of Dukovany units with validity limited by results of the review after 20 years of operation ". Has SUJB imposed any deadline for the fulfillment of these 97 requirements?***

Yes, each of the 97 requirements included in the SUJB Decision No. 197/95 has its deadline for implementation and this is continuously checked. For example during the regulatory assessment of unit readiness for operation after each fuel re-load.

FRANCE 15: Probabilistic safety assessment ("living'' PSA Study)   The report mentions that: "All planned modifications of Dukovany units related to nuclear safety are being evaluated from the aspect of the PSA results, and consequently the priorities of these modifications are being established. The PSA level 1 results have been also used in the preparation of a new emergency procedure". Could the Czech Republic indicate the design deficiencies and the highest priority actions evidenced by the PSA study?

The modifications were reassessed with use of the PSA model in 1995. The assessment of the impact of the planned modifications on the risk of reactor core melting reduction was carried out for all modifications which can be assessed by the PSA-1 model, i.e.:

- modifications proposed by specification sheets;
- modifications resulting from the PSA assessment;
- modifications resulting from re-assessment after 10 years (SAR);
- modifications accepted from the list of IAEA "Safety Issues" for V213 model.

The analyzed modifications can be divided to the following three groups:

- The modifications with impact on reduction of the total CDF;
- The modifications with impact on reduction of the CDF for some initiating events only;
- The modifications with impact on the reliability of the appropriate system only.

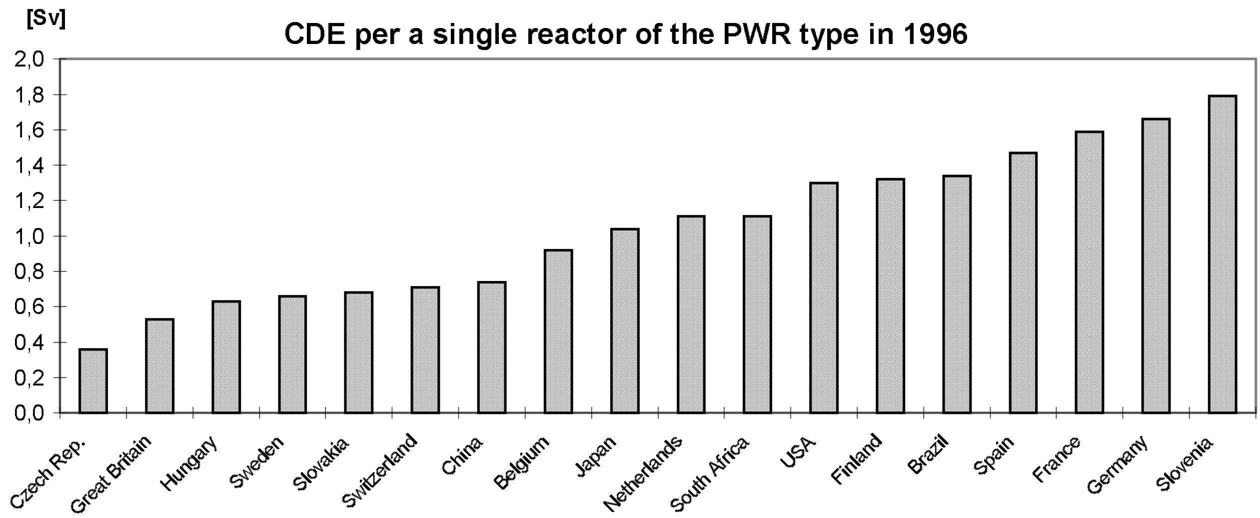
Just for information, we include below the most significant modifications in tabular form. Only those are included that cause reduction of CDF. The results of this assessment are in a good compliance with the levels of seriousness set out according to the deterministic evaluation and used for definitions of priorities of these modifications with impact on safety within the upgrading program.

No.	Modification Designation	CDF reduction (reduced by XX%)	Planned implementation
1.	New disposal solution of the emergency feedwater supply pipelines	34%	1999 - 2001

2.	Equipment installation capable to withstand the impact of environment after piping rupture at the level +14,7m in the adjacent building	33%	1999 - 2004
3.	Installation of pipe whip restraints, supports, protective shields against water jet etc. at the level +14,7m in the intermediate building	28%	1999 - 2004
4.	New emergency operating procedure	22%	1999
5.	Special drainage train from the room of the MCP to the SG room	18%	2002 - 2003
6.	Successful qualification of MCP seals	11%	Done.
7.	Modification of the automatic systems for closure of the quick-response valves TF10 after signal "large accident"	8%	2002 - 2006
8.	Modification of the automatic systems on the discharge train of the emergency feedwater pumps	8%	1999 - 2000
9.	Performance of the reverse switch over of the suction from sumps in containment to LP ECCS tanks	7%	Implemented.
10.	Modification of the RTS and ESFAS signal "main steam header rupture"	7%	Implemented.
11.	Protection of quick-response valves and SGSV against mechanical impacts of piping rupture at the level + 14,7m in service building	4%	1999 - 2004
12.	Installation of the Pressuriser relief valve, the protection against cold overpressure.	2,5%	PORV installed, protection 1999 - 2000.
13.	Measure for large water leak in the turbine hall	2%	Partly implemented.
14.	Modification of the 110 kV substation of reserve power supply	1,2%	No cost effective.
18.	Modifications of the hydroaccumulator system	1,2%	Implemented.

FRANCE 16: Radiation protection of personnel: The report indicates the collective dose observed at Dukovany reactors which is slightly higher than the dose observed in most nuclear countries such as USA, France, Japan and Germany, but no indication is given on the statistical distribution of the individual doses. Could the Czech Republic provide this information, for example using a representative histogram?

Yearly values of the CDE (collective dose equivalent) caught by the Dukovany NPP personnel have been permanently kept at a very low level since commissioning in 1985. If this indicator is matched against the values of the CDE-per-unit of PWR type, it can stated, that in 1996, for example, the Dukovany NPP was occupying the first rung on the world ladder:



In the following tables an outline is available of how the individual dose equivalent (IDE) values are distributed among the Dukovany NPP for the last three years:

Year 1996

IDE Interval [mSv]	**0,00-0,09**	**0,10-0,49**	**0,50-0,99**	**1,00-0,99**	**2,00-4,99**	**5,00-9,99**	**> 10,00**
Number of employees	1505	426	145	154	105	53	6

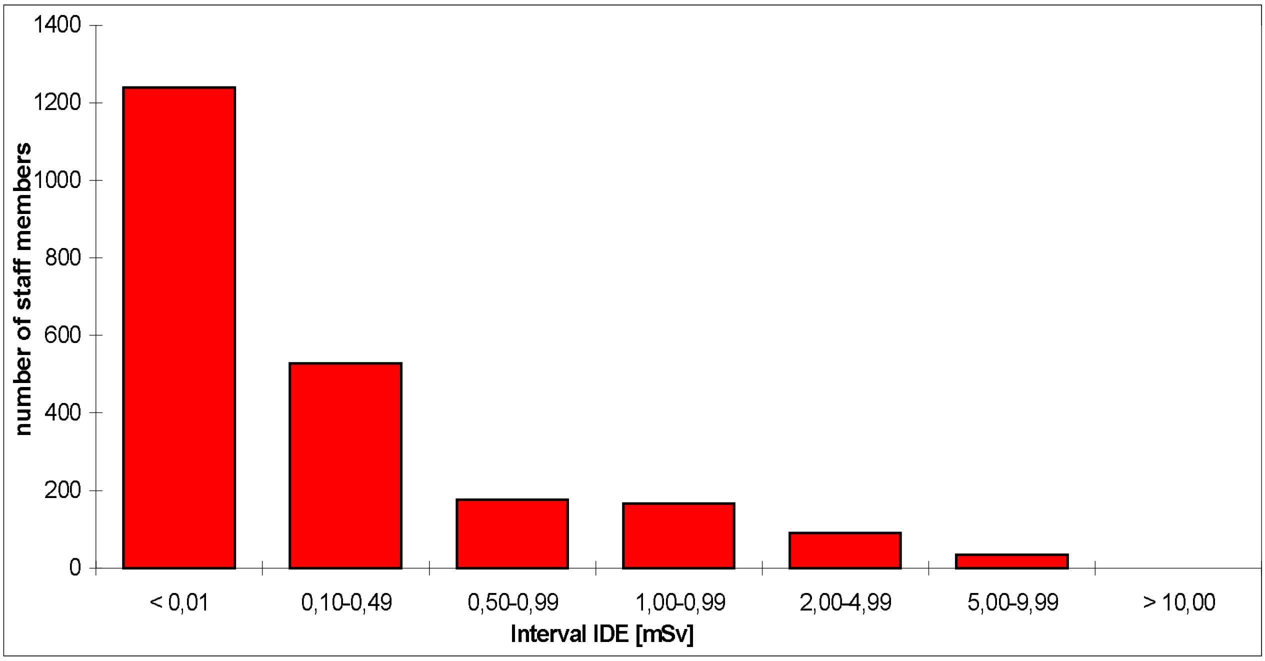
Year 1997

IDE Interval [mSv]	**0,00-0,09**	**0,10-0,49**	**0,50-0,99**	**1,00-1,99**	**2,00-4,99**	**5,00-9,99**	**> 10,00**
Total	1345	494	179	146	135	32	3

Year 1998

IDE Interval[mSv]	**0,00-0,09**	**0,10-0,49**	**0,50-0,99**	**1,00-0,99**	**2,00-4,99**	**5,00-9,99**	**> 10,00**
Total	1240	528	176	167	91	34	0

Below is an illustrative histogram giving the distribution of these IDE values at the Dukovany NPP for the last three years:



FRANCE 17: Radiation monitoring in the vicinity of nuclear installations: The results are given in different annexes. In annex 3, the graphs giving released activities for noble gas, aerosols and iodine do not indicate units (p.28, 29 and 30): so it is necessary to consult the very clear graphs of annex 4 (p.53) to understand that the units used in annex 3 are respectively GBq, MBq and MBq. In future reports, it would be helpful to carefully indicate units so as to facilitate the interpretation of the results.

Comment only, no answer prepared.

FRANCE 18 On-site and off-site emergency plans: Are there computerized support systems to understand the status of the installation, to predict the accident progression and the doses around the plant?

FRANCE 19: What are the criteria to enter in an extraordinary event?

FRANCE 20: Does the Czech Republic perform exercises to test the efficiency of on-site and off-site plans?

FRANCE 21: What are the criteria used to take a decision on measures for protection of the public (sheltering, evacuation, stable iodine tablet distribution)?

Ad a) Yes, these systems are available both to the NPP Operator and the SÚJB ERC. At NPP Dukovany Crisis Center, a SW support systems for radiation events progress prediction such as RTARC (Real Time Accident Release Consequence) or data METEO are available. In addition, a real-time data from on-site and off-site teledosimetric system are available. Similar support systems are being prepared for NPP Temelín. SW for prediction of technology status is being prepared for installation just now.

The Emergency Response Center (ERC) of SÚJB has at its disposal the following computerized support systems:

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

Ad b) The on-site emergency plan of NPP Dukovany contains the part 10/101 “Criteria for classification of the extraordinary events”. There are established the conditions and intervention levels for declaration of the extraordinary events of particular degrees established by Decree No.219/97. The conditions and intervention levels are established from point of view the needs of quick technological status and radiation situation evaluation in the case of extraordinary event origin. Criteria are defined by on-site emergency plan.

Ad c) To largest extent, the on-site emergency plans are being exercised at Dukovany NPP. Plans for drills are prepared for each year. In minimum, one drill a year includes off-site segments of emergency management to practice coordination between these two plans. According to the Law, once per two years a complex exercise has to be organized (level 3). The next one is planned for this fall. This drill will include even the District Authorities and emergency management on national level. The national level of emergency management is involved in international drills, such as INEX 2 or Hexagrant.

Ad d) For criteria used for decision about protective measures for the population sees appendix.

Appendix

Intervention levels for immediate protective actions (urgent measures) to protect the public

Immediate protective actions are always regarded as justifiable if exposition of any individual could result in a damage to health, therefore, immediate protective actions are always implemented if it is expected that the equivalent dose could, within less than two days, for any single person exceed the intervention levels given in table. 1

Table 1

Intervention Levels of Dose for Acute exposure

Organ or tissue	Projected absorbed dose to organ or tissue in less than 2 days (Gy)
Whole body (bone marrow)	1[a)]
Lungs	6
Skin	3
Thyroid	5
Lens of the eye	2
Gonads	1

Note: [a)]The possibility of deterministic effects for doses greater than about 0,1 Gy (delivered for less than 2 days) to the foetus should be taken into account in considering the justification and optimization of actual intervention levels for immediate protective action.

If immediate protective actions could, for a maximum period of 7 days, avert or reduce irradiation in a critical group of inhabitants, to an extent exceeding bottom values of guide values of intervention levels stipulated in table. 2, then the realization of protective actions depends on the extent, feasibility and cost of actions and their eventual consequences; if top levels are exceeded then protective actions are always implemented.

Table 2
Generic intervention levels for urgent protective actions

Protective action	**Extent of doses**	
	Effective doses	**Equivalent doses in individual organs and tissues**
Sheltering and the iodine prophylaxis	5 mSv to 50 mSv	50 mSv to 500 mSv
Evacuation of inhabitants	50 mSv to 500 mSv	500 mSv to 5000 mSv

For realization and evaluation of the extent to which the immediate protective actions are to be taken the guide values of averted dose, listed in Table 3, are applicable.

Table 3
Guidance values of averted dose

Protective action	**Generic intervention level (dose avertable by protective action)**
Sheltering	10 mSv[a)]
Iodine prophylaxis	100 mSv[b)]
Evacuation	100 mSv[c)]

Notes:

a) for a sheltering period not longer than two days,
b) avertable committed equivalent dose to the thyroid due to radio iodine,
c) for evacuation period not longer than one week.

For the decision on relocation the following guidance values of intervention levels are used:

a) for initiation of temporary relocation, an averted effective dose of 30 mSv for a period of one month,

b) for conclusion of temporary relocation, expected effective dose of 10 mSv for a period of one month. If it is found that, during one or two years, effective doses for one month have not fallen down below the intervention level for concluding temporary relocation then permanent relocation must be considered,
c) for permanent relocation, the expected lifetime effective dose of 1 Sv.

FRANCE 22, 17.1. External events taken into account: Could the Czech Republic indicate how earthquake has been taken into account in the initial design of Dukovany NPP?

FRANCE 23: How was it demonstrated that an acceleration of 0.1g, which is the acceleration of the new safe shutdown earthquake, does not lead to safety problems?

During the phases of locality selection, initial design, and actual construction of the NPP with the VVER 440 reactors the problems connected with its seismic resistance were taken into account (in the initial design the acceleration value of 0.06 g was considered).

In its rev. 1, the Pre-operational Safety Analysis Report for NPP Dukovany has shown that in the subjected locality the macro-seismic intensity value of 5.8 MSK will not be exceeded within the horizon of time having 10,000 years in duration, regardless of how conservative approach is entered into calculations. The acceleration of 0.1 g correspondents here to the intensity of 7 MSK-64.

After construction of the Dukovany NPP, the experimental dynamic measurement has been carried out on the reactor building in 1984.

Now, when the Dukovany NPP is being upgraded, the power plant seismic resistance is under revision. Within the works (started in 1995) on qualification of the Dukovany NPP equipment, the individual pieces and constructional objects are being re-qualified to 0.1 g. Results of the qualification itself are documented in the Qualification Protocols.

FRANCE 24, 17.2. Radiological impact of the plant: What are the limitations of public exposure for design basic accidents?

No specific limits for so called “design basic accidents” are set down in current new Czech legislation. General limits (e.g. 1 mSv per year for effective dose) are applied both for normal operation and for design arrangements in relation to reasonably foreseen deviation and accidents.

Of course, severe accidents with impacts exceeding the limits could occur with some small probability, but limitation of such probability is not set down in legislation, neither directly nor indirectly. Accident scenarios, which have to be evaluated are reviewed and checked out in the framework of licensing procedures. Radiological consequences of possible accidents are considered as a part of general siting and design criteria. It is stipulated that for normal operation up to the maximum design basis accident the limits shell not be exceeded and for severe accidents appropriate measures shall be prepared.

FRANCE 25: The report mentions that the safety level of the design of Dukovany and Temelín was assessed by IAEA missions: could the Czech Republic summarize the important safety features (existing or improved) concerning prevention and mitigation?

NPP Dukovany

Positive traits in the VVER 440/213 NPP design include the following:

- Low power density in the core and thus also a high level of safety reserve for variety of accidents and transitory states
- Large volume of coolant in primary circuit and steam generators and related large reserve in time (5-6 hours) for remedial actions in the case of accidents with lost power to SGs (e.g. a blackout or fire in the SG power supply system. It is also advantageous in the case of small LOCAs. Due to large coolant volume the unit is highly sensitive to various abnormal operating conditions.
- Another unique trait of the VVER 440 NPP is in the primary circuit arrangement with six loops. Should a half of the HCČ fail, the power will only drop to 50 %.
- Horizontal SG are used and due to the primary circuit general arrangement the heat can be easier extracted by natural circulation, having a variety of other benefits
- Use of closure valves in the primary circuit loops makes the maintenance works on these loops and shut down SG easier
- By use of two turbines the number of reactor shutdown conditions due to the failed secondary circuit devices could be minimized.
- High level of redundancy in ECCS - design with three trains

The IAEA mission was one of the key parts of the safety-related recommendations incorporated into the MORAVA upgrading program. In its current form the program is now at the phase of implementation (the business aims are being prepared and licensed, as well as the aim of the entire construction). Some of the safety enhancing measures have already been accomplished, other are under accomplishment.

The IAE mission itself has been carried out in late 1995 on the basis of the IAEA-EBP-WWER-03 document being just prepared. The goal is to establish the relevance and importance of the individual problems to the Dukovany NPP, applicable to its condition prevalent in the period of the mission.

In the conclusions and recommendations of the mission the following has been stated:

- The mission has stated that the projects and programs of the NPP Dukovany modernization have been as a whole elaborated in compliance with the international experience. Implementation of the co-ordinated safety-enhancing measures will be a substantial contribution to the plant safety.
- With the design in mind, the mission experts have put under their judgement 74 safety-vital issues. All of them ware clarified with the NPP Dukovany experts and the suggested improvements were judged. Within its measures, the missions have advised further improvements of the suggested measures. Some more measures have also been recommended so that an overall purpose of the IAEA mission was met.
  Since the time of the mission it can be summarized that all the IAEA Safety Issues categorized II and III have been incorporated into the upgrading program by the Dukovany NPP and are now at different stages of completion. As for the operational field, its is

judged at the Dukovany NPP on separate basis and its general standard will be put under verification by the repeated OSART mission in 2001.

The Dukovany NPP intends to bring to life 72 of all 74 recommendations (categorized I, II, and III), doing so step-by-step by 2010.

From the total of 40 measures categorized II 31 shall be accomplished by 2002 and of 8 findings of IAEA (categorized III) all shall be accomplished by the same year. The remaining actions within the category II shall be sorted out together with the action called „Instrumentation and Control System Modernization“. It is scheduled to finish between 2007 and 2009.

- The mission has stated that a large volume of work has been carried out in the field of the units' main components, in particular of the pressurized reactor vessel and primary pipelines.
- The mission has stated that in the field of measurement and control the NPP Dukovany had performed a reliability analysis with its output in the form of a list of modifications to improve the plant protection system. Within the mission other work has been advised to improve the main control rooms (to make them more „human-friendly“ and make the man-machine interface more agreeable). This measure has been included in the upgrading program and in part implemented already.
- The mission has stated that some key results were reached in the issues related to internal risks (fires, floods, pipeline whipping). A progress is apparent in the area of fire risks.
- In the field of emergency analyses the mission have advises to embark on the accident at lower levels of power and in the reactor shutdown conditions under non-stationary and non-homogenous conditions. Since the time of the mission, the analyses have already been carried out.
- The IAEA experts have further recommended implementation of the symptom-supported operating emergency regulations. Also the „Limits and Conditions“ should be improved in line with these recommendations - the symptom-oriented operational regulation of how to do away with the accidents is about to be finished in this year.

It can further be stated that, since the mission performance, a series of measures have already been taken to increase the safety.

NPP Temelín

On the basis of the IAEA missions carried out, the general approach to nuclear safety has been revised against the original Russian VVER 1000 design documentation of 1986. Main attention was paid to reduction of potential emergency conditions and to ways how the consequences could be relieved.

In this area even the changes in the technical design documentation seem to be a major contribution, such as the ones in the I&C system, reactor core, electric and cabling system, causing an increase in the Temelín safety.

Seismic resistance of the Temelín NPP has also been placed under revision with use of the current standards and knowledge in the field of seismic protection.

In addition to these conceptual modifications there were numerous technical changes, increasing the safety of individual system and thus also of the entire power plant.

For example, the following can be stated:

- Addition of the Pressuriser relief valve, lowering substantially the risk of the safety valve hung up in a semi-open state;
- Addition of the catalytic hydrogen combustors into the containment space;
- Reconnection of PSA to the Ist category of the guarded powering;
- Addition of anti-whipping devices onto the steam and feedwater lines in the room A 820;
- Addition of the pipeline cladding, including the first closure valve form the emergency stock tank with boric acid (GA201);
- Extension of the boric water collection system;
- Replacement of the heat insulation aluminum outer finish in the containment for a stainless one, restraining actually hydrogen development after accident;
- Modified manufacturing procedures for steam generator (better characteristics of initial materials, the heat-directional tubes otherwise fixed in their collectors, ...);
- Modified manufacturing procedures for heat exchanger of normal and emergency primary circuit cooling (TQ, two-phase steel);
- Extension of the after-accident monitoring system which is also used for sample taking from the containment;
- Extension of the primary and secondary circuit diagnostic system;
- Extension of the seismic monitoring system;
- Extension of the systems designed to monitor the leaks through the fuel whip cladding - SIPPING;

Concerning the office background, the FSAR has been sorted out according to RG 1.70 and the limits and conditions have been made in line with the NUREG 1431. There is another significant gain in the field of operational safety: Symptom-oriented emergency operational procedures have been compiled, being in full compliance with the recommendations and unit control concepts in the Western power plants.

These symptom-oriented emergency operational procedures made and used for emergency situations represent the most significant benefit in the field of containment of the emergency consequences.

Concerning the office an paperwork background, let’s state the work on the FSAR according to RG 1.70 and creation of the limits and conditions in line with NUREG 1431. There is another significant gain in the field of operational safety: Symptom-oriented emergency operational procedures have been compiled, being in full compliance with the recommendations and unit control concepts in the Western power plants. In this area even the changes in the technical design documentation seem to be a major contribution, such as the ones in the I&C system, reactor core, electric system, causing an increase in the Temelín NPP safety. Besides this, there ware some minor technical changes (e.g. a relief pressuriser valve added, catalytic hydrogen combustors installed in containment), having increased the safety of individual systems and thus also of the entire power plant.

FRANCE 26: What is the new technology for I&C at Temelin and the regulatory approval process? Could the Czech Republic also add some elements concerning testing and maintenance?

Ad a)

The Temelín I&C is a complex set of computer-based systems, using four different computer technologies developed by the Westinghouse Electric Company:

- The standard Westinghouse Eagle Family of I&C equipment, utilized within the Primary Reactor Protection System, Post Accident Monitoring System and safety-related Reactor Control and Limitation System;
- A new "diverse" technology (Motorola processors etc.), applied for the Diverse Protection System implementation (including the Diverse Monitoring Systems);
- The Westinghouse Distributed Processing Family of I&C equipment, used to implement the Plant Control System, the Turbine Control System and the In-core Instrumentation System (being non-safety systems);
- The Westinghouse Workstation Family technology applied within the non-safety Unit Information System.

The reactor protection system of the Temelín NPP consists of the Primary Reactor Protection System (PRPS), the Diverse Protection System (DPS) and the Non-Programmable Logic (NPL).

The digital, software-based PRPS provides principal plant protection at occurrence of any of the postulated DBEs, on a highly conservative level. This protection is performed through initiation of reactor trips, actuation of the plant ESF and monitoring of complete set of plant variables. The safety tasks of the second digital protection system - the DPS - include automatic initiation of reduced set of reactor trip functions, actuation and / or control of selected ESF component and diverse monitoring of minimum sufficient sets of plant variables.

The DPS is implemented to reduce potential unavailability of reactor trip and ESF actuation per demand due to a postulated Common Mode Failure (CMF) within the programmable portions of the PRPS. In that situation the DPS - and the diverse manual controls - enable the operating staff to shut down the plant to both hot shutdown and cold shutdown conditions, to monitor a cold shutdown condition and perform necessary remedial actions. There are no identical hardware or software modules used in both systems. These facts are sufficient to claim the DPS vs. the PRPS "diversity". The DPS by itself provides protection at those DBEs, which frequency is expected to be above 10^{-3} per year. For protective actions initiated by the DPS, relaxed acceptance criteria (in comparison with the PRPS ones) have been permitted.

The NPLs are more or less only special non-digital components of the Temelín NPP Engineered Safety Features Actuation System (ESFAS). Their main function is to increase the priority of the ESF component commands from both hierarchically equivalent protection systems and to perform some parts of actuation logic algorithms. In addition, diverse manual controls and indications interface directly to the NPLs.

There are three independent divisions of the PRPS, the DPS and the NPL, and at normal operating condition the standard voting logic used within both protection systems is two-out-of-three. The capacity of the Temelín ESF system is 3*100%, i.e. in principle only one from three existing ESF divisions is capable to mitigate consequences of the DBEs.

Regulatory approval process of the Temelín I&C is guided by the Standard Review Plan (NUREG 800), which was transformed into a database of licensing issues, with basic structure (i.e. sections) corresponding to standard structure (sub-chapters) of the SAR. Two specific database sections have been added to adequately cover assessment of protection diversity (i.e. the implementation of the DPS and the NPLs).

The database issues are organized in three hierarchical levels. For each issue the result of its evaluation is provided in the database, together with references to applicable standards and guidelines, to sources of information, etc. Moreover, also all SÚJB formal Requests for Additional Information (RAI) associated with the issue, Westinghouse responses to those RAIs and their evaluations are included there.

The final Safety Evaluation Reports summarize main conclusions of evaluation provided in the database discussed above, and includes brief description of evaluated I&C system.

Ad b)

The I&C system maintenance and testing is performed by the power plant operating personnel in full operation. Due to the I&C system internal diagnostics a wide spectrum of malfunctions cane be automatically detected without setting the system out of operation.

In the case of safety systems, the periodic testing is entrusted to the operating personnel doing so with use of the so-called Autotester. The I&C systems will be maintained according to the recommendations and operating manuals handed over by Westinghouse as contractor.

In addition to routine maintenance of I&C (mostly consisting of replacement of defective system modules, damaged sensors, etc.), the personnel can itself repair some simple modules (such as Analog/Digital converters). It is currently assumed, that all I&C software modification will by done by manufacturer (i.e. WELCO); the role of the NPP personnel will be confined here to mere software-holding EPROM replacement and similar activities.

FRANCE 27, paragraph 14.1.2 Limits and Conditions for safe operation: The report states that: "The limits and conditions for safe operation underwent also a re-evaluation grounded on the results of probabilistic safety assessment". In which way were the limits and conditions for safe operation modified by the results of the probabilistic safety assessment? Which parameters were modified: allowed outage times (AOT), tests periodicity, others?

Limits and Conditions reassessment for NPP Dukovany has been carried out by means of the PSA-1 and Risk Monitor.

On the one hand, the Acceptable Outage Times (OAT) of the individual facilities modeled on PSA came under judgement, and a contribution of some other components unavailable so far in the Limits and Conditions (like the TF14W01, TF15W01 exchanger) to the risk have been appraised on the other hand. After mutual discussions between NPP Dukovany and SÚJB, the AOTs were altered for some facilities, some new components were incorporated into Limits and Conditions, and some of the LPs were modified (more restrictive demands on 1st p. for ZN II - e.g. 48 hours for the 6 kV switchgear).

The works on periodicity assessment of testing intervals of some selected facilities with use of PSA are going further on.

FRANCE 28, paragraph 14.1.6 Operational lessons learned approach, Operational lessons at nuclear power plant Dukovany: The report indicates that: "The Plant Failure Commission at its regular sittings confirms completeness of the investigations into causes of safety related events and takes measure to remove them". ... "The significant events are discussed also by the Technical and Safety Commission in the presence of plant top management ". Who are members of these two Commissions? Does the SUJB take part in this process and how? Can a site inspector ask for a meeting of the Technical Safety Commission prior to a plant start-up after a significant event?

Plant Failure Commission (PFC):

There are permanent members of the Plant Failure Commission, in particular the heads of specialized departments of NPP Dukovany (Event Investigation, Quality Control, Technical Quality Assurance and Inspections, Technical Development, Waste Management, Nuclear Safety, Radiation Safety, Operation Management, Overhaul Management, Primary and Secondary, Section Administrator, Control and Measurements System Administrator, Electric Administrator).

Moreover, the shift engineers, representative of ČEZ, a.s. Headquarters and of the Temelín NPP are being invited the meetings.

As required, representatives of the other specialized sections or contractor organizations can be invited to the discussions aimed at the events relevant to them for clarification of problems at equipment, human failures, etc.

The Deputy Director heads the Plant Failure Commission for Operations, the head of the Event Investigation Section occupies the post of secretary.

Members of the Technical Safety Group (TSG):

The TSG consists of the following members: Nuclear Safety Deputy Director, Operational Deputy Director, Maintenance Deputy Director, heads of the specialized sections for Nuclear Safety, Quality Control, Radiation Safety, Engineering and Technical Services, Technical Inspections, Primary and Secondary Part Administration, Operational Management, Reactor Physics, department managers for Control and Management System, and for Electric Systems.
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How is the SÚJB's involved in this system?

Every month, after every Plant Failure Commission session, the SÚJB inspection follows where the events sorted out at this session are discussed pursuant to the Atomic Act. (the ways how they are investigated and evaluated, how the causes are analyzed, what remedial measures are taken are under control of INES). As required by the SÚJB, some power plant specialists may be invited to these discussions in order to explain there the professional aspects of the issues under discussion.

Furthermore, the SÚJB is taking part in every extraordinary Plant Failure Commission being called up on occurrence of a safety-relevant event (like in case of action of the "HO-1", "HO-

2" emergency protections, on failure to meet the Limits and Conditions, on loss of natural circulation without recovery by 1 hour, or also upon an INES event of the lever 2 or higher) as it is stated in the internal document of NPP Dukovany (Directive 09/101). A SÚJB representative has the right to actively intervene with discussions of the extraordinary Plant Failure Commission.

The events under discussions by the extraordinary Plant Failure Commission must be submitted to regular session of this Commission not later than within 30 days.

Has the SÚJB right to request the TSG to be called up prior to a unit start-up following a significant event?

The TSG acts against NPP Dukovany as its counseling body only, the NPP Dukovany technical director calls up the extraordinary sessions. In line with the Atomic Act, the SÚJB inspector is entitled to suspend the unit start-up process if some conditions which has been duly ordered by the Office, were not met by the operator, etc.