Answers to Questions and Comments Raised by Germany

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

GERMANY 1, Page 12, Chapter 1.1.2.2: Safety Issues for WWER-1000/W-320 NPP are compiled in /IAEA-EBP-WWER-05/. Are all major recommendations (Category II to IV) of the IAEA - Extra Budgetary Programme covered by these programmes? What is the schedule for the planned modernization of the Dukovany NPP?

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 summarized that all of the „Safety Issues IAEA“, categorized 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 (categorized 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 categorized II, 31 will be accomplished by 2002 and as for the 8 findings of IAEA in the category III, all will be realized 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 (I), AA4 ((I), AA7 (II), AA12 (I), AA13 (I), AA14 (I), AA15 (I), IH8 (I), EH3 (III)	24	28	11	11	2

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	-	-

Temelín

The extent to which the recommendations from the IAEA-EBP-WWER-05 Report had been implemented was assessed by the specialized Committee of IAEA in March 1996. In summary, it can be stated that most of these recommendations had been fulfilled at that time already or were then in an advanced stage of realization, this being documented in the Summary Report from this IAEA mission called “Reviews of WWER-1000 Safety Issues Resolution at Temelín NPP, WWER-SC-171“.

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 center	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 hazard 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 pressurized 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	Surveillance programme	addressed
PO3	System of communications	addressed
RP1	Radiation protection and monitoring	addressed
Tr1	Training programs	addressed
EP1	Emergency center	addressed

In order to evaluate the state of fulfillment of the recommendations under IAEA-EBP-WWER-05, the NPP Temelín has entered a contract with an engineering/consulting company. The purpose is the detailed analysis of the works done so far, with the goal to timely identify any potential drawbacks in the existing solution.

In general

In the Appendix to the National Report, the Summary Reports of the IAEA missions, the ones that have evaluated the „Safety Issues“ implementation in NPP Dukovany and NPP Temelín, are referred to. They can be made available through Mr. Šváb, „liaison officer“ of SÚJB.

GERMANY 2, Page 21, chapter 2.1.2: Does the regulatory authority involve experts from technical safety organizations (e.g. NRI Rez) in regulatory inspection and safety assessment of nuclear installations? Do these experts also perform activities for the operator of the same nuclear installation?

The SÚJB is checking that the Atomic Act and other related regulations based on this Law are observed. For checking purposes, the SÚJB has set apart its nuclear safety inspectors and radiation protection inspectors. The Office chairman appoints them. It means that just these inspectors and nobody else are in the position to check that the provisions of the above Law and executive regulations are kept by the law-defined persons (individuals and corporate legal entities, whether or not they obey the license in its subject and scope, including the stipulated conditions. In order to ensure these activities, the SÚJB is also making use of the external experts hired from other organizations, though doing so for the technical support only.

It means in practice, for instance, that the SÚJB inspectors carry out all inspections only. While evaluating nuclear (radiation) safety, the SÚJB is, where necessary, assigning the various analyses, calculations, feasibility studies, and the like to the external experts, using them there as of the underlying documents to support its own evaluation. Consequently, the final resolution is left again to the SÚJB inspectors only. The contracts for feasibility studies are awarded to those experts only, who are not involved in the same activity for the licensee (or applicant). In the area of radiation protection, they are mainly the experts from the State Institute of Radiation Protection, being an organization funded from the state revenue and administered straight by SÚJB. In the field of nuclear safety a variety of experts is what matters here. Concerning the Nuclear Research Institute, Řež (NRI Řež) stated in the question, it has established a specialized department with its responsibility to provide the SÚJB with technical support The department is set apart to be exclusively engaged in the SÚJB contracts. To include an example here, let us mention that the experts from this department have already done within one SÚJB order an independent assessment of the safety analyses (chapter 15 under US NRC Regulatory Guide 1.70) of the Power Plant Temelín innovated design. As an independent technical support, the specialists from universities or various consulting and engineering organizations are further employed.

GERMANY 3, Page 62, chapter 9.13: Are there plans to extend the existing ageing programme on main components to e.g. instrumentation and control, electrical equipment, structural components including the pre-stressed cables as well as documentation?

The equipment of the ČEZ. a.s. nuclear power plants is under systematic reviews with focus on service life of their key components significant for nuclear safety, including systems of cabling. Within the comprehensive program of service life estimation a subprogram has been opened designed to estimate the life of cables in the Dukovany and Temelín NPPs, managed from the level of the ČEZ, a.s. Headquarters. Within this subprogram, key attention is paid to the cable system affecting the nuclear safety.

Within the project called Implementation of the Cable System Ageing Management in the Czech NPPs the cable have been picked for monitoring of such ageing and methods of measurement and evaluation have been devised.

The results of this above program aimed at the cable life span will be used for objective estimation of service life for these cable systems up to the moment when they are still reliable and safe. The entire program has been documented in details and reviewed from time to time.

By the „Ageing Management Rules“ just issued, the existing program of ageing management of key components in the NPP Dukovany has been extended to cover even the measurement and control system components, electric ones, and those belonging to building structures. The control life span of all components is what matters here, with an influence on the nuclear safety.

In the NPP Temelín, a preliminary stage of engineering, technical, and organizational works is in run now, designed to make a system of service life monitoring of the individual containment sections.

The works are concentrated on:

a) Technical underlying materials and documentation;
b) Selection of consulting organizations.

In co-operation with the architect designer organization, the elements (parts) of the containment to be monitored are being figured out within the limits of operational and inspection works, together with scope and periodicity of monitoring.

The discussions have been opened with the Faculty of Civil Engineering of ČVUT, Praha for definition of the goals of solution and methodology of life-time monitoring on the containment sections (concrete, pre-tensioning cables, concrete reinforcement, hermetic lining, etc.)

The Department of Civil Mechanics of the above ČVUT Faculty has its share on the projects related to NPP inspection and monitoring within the RILEM, OECD, and EU programs.

GERMANY 4, Page 79, chapter 11.1.2: Will on-site and off-site emergency plans be prepared for the Temelín NPP before the start of plant commissioning? What is the reference doses for initiating emergency measures?

Ad a) It is stipulated unambiguously in the Atomic Act that the NPP Temelín on-site Emergency Plan must be prepared in compliance with the requirements set out in the regulation (SÚJB Regulation No. 219/1997 Coll.). The interface with the off-site emergency plans must be prior to that discussed with the relevant District Offices. The SÚJB should approve the on-site emergency plan, still before the Office issues its permit to feed the nuclear fuel into the reactor.

The Temelín NPP off-site Emergency Plan must be prepared within the responsibilities of the Czech Republic State Administration, in particular by its District Offices at České Budějovice, Písek, Tábor, Strakonice, and Prachatice, i.e. the districts whose territories are within reach of the Temelín NPP emergency planning zone.

Ad b) The values of reference doses authorizing the protective measures that can be initiated once the values are reached are given in the SÚJB Regulation 184/1997 Coll., being in compliance with the Basic Safety Standards IAEA, Safety Series No. 115-I, and ICRP recommendation No. 63. Basic guidance effective dose values on which the emergency measures can be declared (5 to 50 mSv for sheltering and iodine prophylaxis and 50 to 500 mSv for wide evacuation) are supplemented to include the values of averted effective doses (10 mSv/2 days for sheltering and 100 mSv for iodine prophylaxis, and 100 mSv/week for evacuation). For details to response levels for urgent population protection measures see the 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.

GERMANY 5, Page 97, chapter 13.1.2, Annex 1: Which codes and standards are used for the ”Modernization Programme for the Dukovany NPP”?

The design of the Dukovany Nuclear Power Plant corresponds to the unified design documentation of VVER 440/213 with respect to OPB 72 (Russian Standard - General Safety Rules), the Decree of the former ČSKAE No. 2/1978 on nuclear safety in designing, licensing, and actual execution of the constructions with a nuclear energetic facility and the SÚJB Regulation No. 214/1997 on quality assurance during the activities connected with utilization of nuclear power and the activities resulting potentially in irradiation an on stipulation of the criteria for classification and division of the selected facilities with regard to safety. As the auxiliary materials, the IAEA Guides are used, mainly then the one of 50-C-D (Rev. 1/1998) Code of Safety of NPPs: Designs and Instructions of the 50-SG-DX series.

In its part on the safety enhancing measures, the upgrading program has come out not only of the conclusions reached somewhere beyond the Operational Safety Analysis Report but also of those from the external audit and IAEA mission. For the external audit, the evaluation criteria have been proposed, based on the IAEA standards and codes. It should be noted here, that there are some particular cases (pressurized vessel integrity, assessment of measurement and control systems) where other widely recognized standards (IEE, ASME, etc.) have also been used. Also the IAEA mission and its supporting PHARE projects ware based on the NUSS series and other documents of IAEA. More details will follow further on.

Generally spoken, these activities represent the periodic safety reviews in their execution, being adapted to our local legislation and specific conditions of NPP Dukovany construction and operation. A considerable amount of work has been done on detailed analysis of the revealed problems and drawbacks.

The outcomes of the comprehensive evaluation became a starting point to the "MORAVA" upgrading program. In summery it can be stated that this program has been prepared in line with the procedures widely applied in the current safety routines of the EU countries.

GERMANY 6, Pages 98, 101, Chapters 13.1.2 and 3: What kind of on-site accident management measures are under consideration or already in place at the Dukovany and Temelín NPP to prevent severe accidents or mitigate their consequences (e.g. containment venting, bleed+feed, additional emergency power supply)?

NPP Dukovany

There at the NPP Dukovany the measures have been and still are being step-by-step prepared and implemented which help prevent the severe accidents or make their consequences easier. These modifications are prepared and implemented in compliance with the recommendations resulted from the international evaluation of the nuclear power units of the type VVER 440/213 and with the conclusions of the analyses carried out.

Mainly the high-power pipelines within the feedwater and steam system must be made more resistant. A systematic risk analysis has been carried out with impact on the safety significant assemblies and components of NPP, resulting in proposal and subsequently in implementation of these measures.

The pipelines within the independent system designed to deliver the super-emergency feedwater into the SG had to be re-routed so that they would not run parallel with the routes of the high-power pipelines. Thus neither the pipeline integrity nor the reliable serviceability of fittings within this system can be affected by broken steam line or feedwater line.

Moreover the reconstruction of the DG stations will be embarked on in the electrical sections to increase the reliability of emergency electric powering.

At the same time a pre-design modifications are prepared with the goal to extract the surplus heat by means of an equipment on the secondary circuit - during the emergency condition - connected with integration of the signal to close the isolating valves (MSIVs) with unavailability of the standard cool-down equipment as its consequence. In particular, a feed and bleed mode of operation in the secondary circuit is what matters here, if the MSIVs are closed.

Furthermore, the project of equipment qualification is being run, where all the equipment is verified, being subject to the qualification requirements on the capability to work in accident of after-accident conditions.

A modification has been carried out, making it possible to run the unit in this „feed and bleed“ mode on its primary circuit if the feedwater supply into the SG is totally lost. The „feed and bleed“ procedure was implemented into the EOP.

Earlier already, the EOP has been extended to include the procedure for renewal of electricity supply into the unit in the case of a „black out“ accident with use of the power from neighbouring reactor-driven units or from a nearby water power plant of Dalešice.

At present a realization of the technical measure is being prepared in the system of coolant drainage from the collection pits in the SG boxes and this will exclude the risk of their clogging with fragments from the torn off pipe insulating layers in the case of LOCA accident.

An action designed to deaerate the primary circuit is being prepared that will, in case of the bubbles under the reactor lid, make it possible to drain them out and provide for heat extraction from the primary to secondary circuit.

In 1998 the following analyses of severe accidents were finished:

- Project of PHARE 4.2.7a „Beyond DBA Analysis“
- Study of PSA-2 in a limited scope, being an analogy of IPE (Internal Plant Evaluation), prepared for the American NPPs within the grant awarded by the US government. This study has been elaborated by the American company of SAIC and its subcontractor of NRI Řež.

In the last year, this PSA-2 study has been updated in its limited extent in order that it would comply with the results of the current study of PSA-1. These research projects have further underlined the rationale supporting the preventive measures that has already been taken or being prepared and have thrown their light on the possibility to implement some relieving measures. The SAMG, when processed, will mainly be based on the outcomes of this PSA-2 study and from conclusions reached within the PHARE 4.2.7a project. There are some important measured on schedule designed to improve the capability of the Main Control Room to be further manned after a severe accident. Moreover, an Accident Management Center is in operation since 1996, performing the role of the technical support center to a sufficient extent.

At present the projects of PHARE 2.06/94 „Analysis of the Need and Alternatives for Filtered Venting of the Containment“ a PH 2.07/94 „Handling of Hydrogen in Containment during Severe Accident“.

NPP Temelín

The measures in the field of accident management are broken down into two groups at the Temelín NPP:

1) The measures to prevent the emergency conditions from escalation into the domain of severe accident.
2) The measure to relieve the severe accident consequences (fuel melting in the core).

Ad 1)
Here will come the design and operational-based measures, preventing in rise the conditions that might escalate into the domain of severe accidents or confine the duration of time for which these conditions are pending.. At the level of the design documentation the goal is to make the NPP Temelín design more resistant. As for the operational readiness, the measures are in question to be implemented within the scope of the NPP Temelín Emergency Regulations. The list of this condition is given below:

- To lower the probability of ATWS events - a Diverse Protection System has been added to the Protection System Design), to shut down the reactor in case of a failure in the PRPS.
- To lower the probability of the „station blackout status“ - two other independent DGs have been added to the Electric Power Supply Design (concept of the 3 x 100% back up, DG is powering the guarding systems), vital for safety. These DGs constitute an independent power supply feeding providing electricity to safety systems of the Temelín NPP. In the

event of a „station blackout“ it is possible to arrange for powering from the Lipno hydroelectric power plant.

- To arrange for heat extraction in the case of the core emergency cooling lost - an such a case the method of „feed and bleed“ is used within the limits of EOPs to extract the heat from the core. For this mode the HT systems of the core emergency cooling are used together with the emergency extraction of the of the steam/gas mixture, or also the pressuriser relieve valve.

Ad 2)

Mainly the operational measures belong to this group for severe accident management. Within these severe accident management procedures (SAMGs - the work on them will be started in the course of this year), the following measures are anticipated to be taken:

- The surface will be enlarged to catch up the spilled melt after failure of the reactor pressurized vessel bottom - by opening the hermetic closure from the reactor shaft.
- Control of containment showering to prevent combustion and potential hydrogen explosions in the containment.
- Pressure reduction in the primary circuit before failure of the reactor pressurized vessel to prevent direct rise of containment temperature.

GERMANY 7, Page 99, Chapter 13.1.2: Which measures are under consideration or already in place at the Dukovany NPP to minimize the possibility of human error and to improve the man-machine interface?

In line with the new international methodologies, the human factor must be closely watched and evaluated. On the basis of this evaluation the measures have been taken and accomplished which will contribute a lot to this human factor reduction at the NPP Dukovany (reduction to one half). They are the measures of both types: organizational and technical.

The following are examples of the organizational ones:

- Re-compilation of the operational documentation (procedures and parameters must be revised mainly for increased factor of utilization);
- In this year a transition will be accomplished from the event-oriented emergency procedure to the symptom-oriented one for which the personnel has been consistently prepared and exercised;
- Enhancement of training procedures (specialized preparation of lecturers, improvement of event feedback, etc.);
- Extension of the „Full-scope Simulator“ training“ to include the exercises on the newly installed Multi-purpose Simulator - including the faults from LF in the training scenarios.
- Establishment of a new post of Safety Engineer for every shift (apart from other he will perform an independent checks that the limits and conditions are really kept, as well as the operational regulations, that the training is prepared within his shift, being responsible for controlled liquidation of emergencies on the inflicted unit, including the inspection of critical functions, remains in communication with the Technical Support Center commander, etc.).

The following, for example can be included in among the technical measures:
- At the MCR, and also as required by the MCR personnel, some indicators and controls have been moved for better ergonomy. The obsolete secondary meters are replaced by the new advanced ones (like the plotters in combination with the digital display units, etc);
- The System of Operator Support has been installed in the MCR, being significantly helpful to the on-duty MCR personnel. This system, for example, facilitates the following:

  - To activate the full archive of parameters
  - To compile whatever diagrams catching the parameter course
  - Various calculations (critical conditions,
  - 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 system displays is step-by-step being replaced by the advanced system of SCORPIO, developed within the framework 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.

GERMANY 8, Page 100, Chapter 13.1.3: For the completion of the Temelín NPP technical equipment from different countries, designed and manufactured at different times, have to be integrated. Which codes and standards were finally used for the design and construction of the Temelín NPP? Is the relevant safety documentation available for both the operator and the regulator to perform integrated safety analysis and independent reviews?

Which codes and standards were finally used for the design and construction of the Temelín NPP?

The equipment supplied to the Temelín NPP is being designed, engineered, factory tested, inspected, delivered, stored, installed, tested, and operated in accordance with the applicable codes and notices of the Czech Republic, standards and other documentation (OA/QC, attached technical documentation, installation instructions, user manuals, operating manuals, etc.). The codes, notices of the Czech and other documentation (hereinafter referred to as „standards“ only) are specified in the contract covering supplies of equipment and services.

The following priorities are applicable for construction of the Temelín NPP:

- Priority No. 1 - Czech standards (including the IEC, EN, ISO, etc. adopted in the Czech Republic)
- Priority No. 2 - International standards
- Priority No. 3 - National standards (applicable in the contractor’s country).

Codes and standards must be unconditionally met.

Obligatory condition was not to affect other parts of the design (design's compatibility)

In addition to this State Regulatory Body (SÚJB) it is requested that delivered systems and/or components has to be "licensable" in the country of origin, i.e. has to meet the national codes and standards of the supplier’s country.

Safety assurance for safety-related items (e.g. fuel, I&C) has to be demonstrated by submitting complete documentation (as a Supplement to the Safety Analysis Report and Topical Reports). For example, for new Westinghouse fuel this documentation includes:

1) design compatibility with other components and parts taking into account the existing (original) materials, moderator (water chemistry) especially from the standpoint of:
   - thermal hydraulic properties - vibration, hydraulic resistance, CHF correlation, fuel rod bowing, effect of spacing grids, pressure losses,
   - mechanic properties - rigidity, cyclic fatigue, wear, cladding abrasion, deformation by external forces (load during LOCA and seismic events), kinetics of control assemblies drop,
   - chemical properties - corrosion, hydrating,
   - neutron-physical properties - peaking factors, influence of different enrichment, water-uranium ratio, etc.; shutdown reactivity margin; stability; maximum speed of the reactivity insertion, both calculated and experimental (especially for non-active tests area).

2) design reliability and safety-related influence by proving that:

- fuel design parameters will not be exceeded,
- fuel cooling will be ensured,
- core design neutron parameters will be met for normal and abnormal operation and emergency conditions (as defined in the Decree N 2/1976 Coll. and/or in 10CF50 App.A, or in equally binding Guidelines of the fuel manufacturer's country).

Is the relevant safety documentation available for both the operator and the regulator to perform integrated safety analysis and independent reviews?

For the purpose of Temelín licensing review a specific set of licensing documentation was submitted to SÚJB and it is continuously updated in accordance with review progress, SÚJB requirements and the I&C design development.
This set consists of :

- Amendment of Preliminary Safety Analysis Report (and Final Safety Analysis Report in future)
- So-called Topical Reports, dedicated to e.g.:
  - Tests results and their evaluation (e.g. Fuel Assembly Hydraulic Test Report, Fuel Assembly and Rod Control Cluster Assembly Mechanical Tests Report, Critical Heat Flux Tests and WX1 CHF Correlation values)
  - Detailed analysis and methodology (e.g. Temelín LOCA Analysis Methodology Report, Westinghouse Reload Safety Evaluation Technology, Core Physics Methodology Report)
  - Analytical Models (e.g. Improved Analytical Models used in Westinghouse Fuel Rod Design Computations, Revised PAD Code Thermal Safety Model)
  - Operational experiences (e.g. Summary of Westinghouse Fuel Performance &Experience)
  - Design basis evaluation (e.g. Core Components Final Mechanical Design Report, Fuel Assembly Final Mechanical Design Report)
  - Individual I&C systems important to safety (i.e. Primary Reactor Protection System, Diverse Protection System, Non-Programmable Logic, Post Accident Monitoring System and Reactor Control and Limitation System),
  - Certain general activities (e.g. Quality Assurance Plan, V&V Plan Topical Report, V&V Summary Results Topical Report, Equipment Qualification Methodology Topical Reports, FBA/FMEA Topical Report, I&C Reliability Analysis Topical Report). The V&V Plan Topical Report includes adequate information about plans for all activities connected with the SW development process.

In addition, the detail design documentation, functional and qualification testing plans and result documents etc. are available to SÚJB on the NPP site or in the I&C manufacturer‘s offices

GERMANY 9, Page 101, Chapter 13.1.3: Which measures are under consideration or already in place at the Temelín NPP to minimize the possibility of human error and to improve the man-machine interface?

One of the main principles used in the Temelín NPP to minimize the errors caused by human factor is a high level of automation in the processes controlling the entire technology, in addition to well-prepared operating personnel. Prior to the power plant commissioning, the operating staff must pass through their training on a full-scope simulator in the area of normal, abnormal, and emergency conditions. The purpose is to attain the knowledge both in technology itself and in the field of man-machine interface.

In the field of operating personnel, the engineering psychology was stressed properly, the requirements of which have to be respected in proposing the man-machine interface and taken as tools minimizing these human mistakes. Currently, the systematic activities are carried out at the TEPP to check and validate there the Main and Emergency Control Room Design.

The Design of both control rooms, main and emergency ones, is, in its concept, a set of controllers and communicators (panels and consoles) combined with digital devices - workstations. The control rooms are designed as entire assemblies of these devices, which, in their arrangement, make use of the individual modes of control and communication in the field of technological processes, doing so in their advantageous and disadvantageous aspects. Thus, a high level of knowledge was attained among the operating personnel, while a clear insight into the state of technologies could be kept. On the basis of this, the operating personnel can make effective decisions when controlling the technology as a whole. Due to the digital equipment, the information can be transmitted to where a shift engineer is working and to the Technical Support Center, providing thus the operating personnel with a technical background in their management of emergency situations and accidents.

At the same time, the operating personnel have implemented some the other measures apart from others, to eliminate any unwanted handling. When controlling a technology, the operators must carry out two actions before the command can be executed. In case of control from a workstation, the operator must select the appropriate actuator at first, and thereafter only either of the commands open, close or start-up, shutdown may follow. Upon controlling a technology from panels and consoles (conventional controllers and communicators), a pre-conditioning push button must be operated simultaneously with an actuator of the relevant piece of equipment. The only exception here is a choice of conventional controllers (such as those to activate an emergency protection and security systems) where the above unwanted handling is prevented by physical barriers (the operator must uncap the actuator prior to action).

In order that the errors could be minimized upon transition form the main control room to the emergency one, the individual working points must have an identical pattern of arrangement at both of these rooms. In this way the problem of operating personal uneasy orientation in the emergency control room has been sorted out.