Gesellschaft für Anlagen-
und Reaktorsicherheit
(GRS) mbH

Executive summary of the results of the in-depth GRS assessment of selected safety issues relating to the Temelin NPP, including supplements

GRS, 25/10/2000

Preliminary remarks

The German-Czech Commission on Nuclear Safety, which was set up in 1990 and is headed by the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) and the Czech nuclear regulatory authority SUJB, deals in particular with safety issues relating to the nuclear power plants lying closest to the German-Czech border, i. e. the Isar 2 NPP and the Temelin 1 NPP (NPP-T). The German side has insistently pressed for detailed information on the safety of NPP-T in order to be able to form its own opinion on the safety status of the plant on the basis of selected safety issues.

National and international studies of VVER-1000-type nuclear power plants performed in the past have already shown up numerous deficiencies and the need for improvement in many areas. The major generic results of these analyses regarding the original design and the operating experience of the VVER-1000/V-320 type - referred to in the following as 'standard VVER-1000' - were jointly presented in a report published by the IAEA in 1996. It numbers 84 safety issues for the VVER-1000, 11 of them in the most safety-relevant category III, where precautions are insufficient and a need for direct improvement exists. Deficiencies relating to category IV, where the level of safety precautions is unacceptable and immediate action is required, were not identified for the VVER-1000 in the international studies.

The BMU has ordered *Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH* to update the general assessment of the safety status of NPP-T with regard to the known deficiencies of the standard VVER-1000 and analyse five selected safety issues in depth. A corresponding order for an in-depth analysis of two further important safety issues was placed by the Bavarian State Government. Accident analyses have been performed by order of the BMU as well as of the Bavarian State Government. With these analyses, seven of the eleven important safety issues indicated in the IAEA report were covered. The aim of these in-depth analyses is to check whether from the German point of view, the solutions provided for NPP-T remedy the respective known deficiencies to a sufficient extent or whether deficiencies continue to exist which have to be eliminated in order to comply with the safety interests of Germany and bordering countries.

The documents needed for the safety analyses were provided on the basis of a German-Czech supplementary agreement on the exchange of information between the Czech nuclear regulatory authority SUJB and the BMU.

The clarification of the technical situation and the assessment by GRS concern underlying concepts and some measures that are planned or have been implemented in this respect. The aim was to check whether and to what extent there still remain any safety-related shortcomings or deficiencies afterwards. What has not been analysed in this context has been the question of how far the safety-related components and systems may be damaged due to an earthquake and whether this may result in accident sequences other than the ones analysed here.

To make statements on licensability was not the task of GRS, nor to review the implementation as specified of the individual measures in the Czech licensing procedure or on plant level. The performance of the licensing procedure and the supervision of the complete fulfilment of the licensing prerequisites is the sole responsibility of the authorities responsible for nuclear safety in the Czech Republic.

Federal Ministry for the Environment, Nature Conservation and Nuclear Safety
October 2000

Part 1

Executive summary of the results of the in-depth GRS analysis of selected safety issues relating to the Temelin NPP

- Taking into account the information available as at September 5, 2000

The BMU and the BStMLU have ordered GRS to continue its analyses performed until 1999 relating to the safety of the NPP-T. The reason was that the information available at the time did not yet allow an clarification of the safety status in all points.

The BMU ordered GRS to perform the following tasks:

– Extended evaluation of the plant-specific information on the NPP-T, especially on those systems and components that deviate from the standard VVER-1000/V-320 design.

– In-depth assessment of sensitive safety issues.

– Assessment of the safety status on the basis of in-depth safety assessments of the safety-related issues identified to be relevant.

– If necessary, derivation of recommendations for a further enhancement of nuclear safety.

The BStMLU ordered GRS to carry out the in-depth assessment of further sensitive safety issues.

The BMU order comprises two work packages:

In work package 1, the unanswered safety-relevant issues are to be clarified, and any possibly remaining safety-related deficiencies or insufficient safety demonstrations are to be identified. These activities are still continuing and will be completed by the end of 2000. They build on studies that GRS and other institutions have already carried out with regard to the NPP-T and other VVER-1000 plants.

In the preliminary analyses of GRS, some sensitive safety issues were identified for the standard VVER-1000 which have shown or may reveal larger deviations from the safety-related requirements of the German regulations. These sensitive safety issues - work package 2 - have in the meantime been analysed in depth for the NPP-T. Including the work done on the order of the BStMLU, they concern:

- the effects of the modified core design on the load behaviour of the reactor (compatibility with the control concept of the plant as a whole),
- the interfaces between the digital safety instrumentation and control (I&C) system and the process-related part of the reactor plant as well as an assessment of the interfaces between the Westinghouse I&C and the Russian I&C,
- the method to demonstrate the progressing neutron embrittlement of the welds and in the base material of the reactor pressure vessel (RPV),
- the protection measures against consequential damage in the case of a 2A-break of main-steam lines on the 28.8-m elevation,
- the control of small loss-of-coolant accidents (LOCAs) including leaks from the primary to the secondary system, namely
    - the completeness (event spectrum) of the accident analyses for a categorisation of the accidents to be considered
    - the quality of the accident analyses concerning small LOCAs including leaks from the primary to the secondary system (break of one and several steam generator tubes).
    - the qualification and reliability of the systems and measures needed to control small LOCAs including the rupture of one or more steam generator tubes
- the ensurance of the capacity of the containment strength by preventing tension losses in the prestressed cables or the failure of the latter, as well as containment leake rate testing, and
- the protection of the plant against external impacts such as aircraft crashes and explosions in the proximity of the NPP.

The detailed GRS reports on the results present for each issue the technical situation that was clarified and the underlying assessment criteria. The assessment criteria used are generally the design-independent safety requirements of the current German nuclear guides and standards and regulatory practice that have to be fulfilled with regard to the different levels of the defence-in-depth concept. The safety concept applied in Germany is described in summary in the "Report under the Convention on Nuclear Safety by the Government of the Federal Republic of Germany for the First Review Meeting in April 1999" on the fulfilment of the obligations of the Convention on Nuclear Safety. Where German safety requirements are not fulfilled, it was assessed whether a safety-related deficiency exists as a result of the deviation and whether the former is - or is planned to be - eliminated or mitigated by compensation measures. International safety practice and other regulations or guidelines of the US-NRC, IAEA, IEC and ICRP were also consulted in supplement. This procedure had been chosen by the BMU for the assessment under the given circumstances because the international state of the art in science and technology is not authoritatively documented in its entirety.

For the assessments of the capacity of the containment strength and the interfaces of the digital safety instrumentation and control system, subcontracts were awarded by GRS to the engineering firm Prof. Eibl+Partner GBR and to ISTec, respectively.

Between December 1999 and June 2000, documents and information that had been reviewed and cleared by SUJB were either handed over for the analyses, or access to them was given. What was provided was mainly extracts from the updated Safety Analysis Report intended for the pre-operational phase (POSAR) and additional written documents or replies in writing to questions posed by GRS. Most of these documents were explained during workshops.

The depth and the procedure of the assessment by GRS were governed by the collaboration of the Czech authority and the degree of detail of the information that was provided. Analyses with the same degree of depth as would be the case in a licensing procedure in Germany were neither possible nor intended.

In the view of GRS, the documentation and information provided proved to be sufficient - with a few exceptions that are indicated - for clarifying the technical situation relevant for the assessment.

Assessments performed by GRS have led to the following conclusions.

- **Assessment of the effects of the modified core design on the load behaviour of the reactor**

For the standard VVER-1000, operating experience and the safety-related assessments have revealed the following weak points of the core design:

– Insufficient effectiveness of power distribution control in the core in the event of xenon oscillations.

– Lacking stability of the fuel assemblies to withstand distortion, with the consequence of longer insertion times of the control elements.

– Insufficient effectiveness of the shutdown rods in certain plant states.

The NPP-T core design was modified on the order of the licensing authority SUJB to eliminate these weak points.

The assessments of GRS with regard to the modified core design led to the following results:

– Changed load behaviour

  The requirements of the German regulations for the safety levels 1 to 4 are fulfilled. The load control concept for the reactor core corresponds to the one applied for Westinghouse plants. The decisive parameter is the adherence to a constant value for the axial offset. Larger axial asymmetric power density distributions can thereby be excluded. The reactivity feedback effects of a fuel temperature increase are negative, as is normal for LWR fuel. The moderator temperature reactivity coefficient, too, is not positive in zero-load hot state, i. e. it is negative for all reactor conditions. This means that the requirements for sufficiently negative reactivity feedback effects in case of a rapid power increase are fulfilled and that the behaviour is inherently safe.

– Fuel assembly and core design

  The assessments performed in this area showed that the fuel assembly and core design fulfil the safety-related design-independent requirements of the German regulations. The technical design of the fuel assemblies and the core rests mainly on the concepts usually applied by Westinghouse and on their operating experience. This means that the known weak points of the fuel assembly and core design of standard VVER-1000 reactors are eliminated.

- **Assessment of the interfaces of the digital safety instrumentation and control (I&C) system**

The inadequate reliability of the safety instrumentation and control system was one deficiency of standard VVER-1000 plants. For this reason the original safety I&C system in the NPP-T was fully exchanged for a digital I&C system made by Westinghouse.
Safety I&C becomes operative in the event of abnormal operation, design basis accidents, and beyond-design-basis accidents. Safety I&C therefore has to be designed in line with the requirements of safety levels 2, 3 and 4 of the defence-in-depth concept. The assessment considers the following aspects of the digital I&C system at NPP-T:

– redundancies and diversity in the safety I&C system,

– interfaces with I&C systems made in Russia,

– technical interfaces, and

– interfaces with the electric power supply system.

No Russian-made I&C systems are in use in safety-relevant areas. Therefore there are no problems with regard to the interfaces with these I&C systems.

The basic design concept and the interfaces of the safety I&C system essentially fulfil the safety-related requirements of the current German regulations. The following deviations were found:

– In extremely rare cases, the failure combination of a random failure and a maintenance outage cannot be controlled by that part of the reactor protection system which effects the control of the safety system. According to the German regulations, this failure combination has to be considered in the design of the reactor protection system, independent of the frequency of occurrence.

  The IAEA also recommends fulfilment of the above-mentioned failure combination. However, the restrictions in the control of this failure combination for different occurrence frequencies are also in line with US-NRC practice.

– In the areas of the measured-data acquisition system and the diesel load logic (NPL) of the reactor protection system, there is no equipment-related diversity for the actuation of the active safety systems of the reactor protection system. According to the German regulations, there have to be provisions in design of the reactor protection system to prevent systematic failures.

The IAEA also recommends diversity in the area of safety I&C as a measure to prevent systematic failures.

– The capacity of the batteries does not fulfil the minimum discharge rate of 2 hours required by the German regulations. Although a demand for an increase of battery capacity cannot be derived from the international regulations, it should be checked whether it is possible to increase battery capacity, particularly in order to achieve greater flexibility with regard to accident management measures.[1]

For the rest, the information and documents provided allow the statement to be made that the design principles and the quality assurance measures implemented are suitable for the design of the interfaces of the digital safety I&C system of NPP-T.

- **Assessment of the method to demonstrate the progressing neutron embrittlement**

The design of the VVER-1000/V-320 shows an increased susceptibility to neutron embrittlement in the beltline weld of the reactor pressure vessel. The reason for this is an increased nickel content of the material, added by the high neutron flux due to the design of the narrow water gap between reactor core and reactor pressure vessel.

Earlier safety assessments of VVER-1000 plants have identified the insufficient database for predictions about neutron embrittlement and the lacking representativeness of the surveillance sample programme with regard to neutron flux and irradiation temperature as safety deficiencies.

The changes in the properties of the materials of the reactor pressure vessel due to neutron irradiation have to be considered in the integrity demonstrations that have to be provided for the requirements on all safety levels.

In this respect, the German regulations and German practice demand the following:

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[1] see also page 16

- a limit on the end-of-life fluence to which the materials are exposed to values < 10^{19} cm^{-2} (energy ≥ 1 MeV),
- the ensurance of a tough material condition with consideration of the expected changes in material properties over the entire operating lifetime,
- a surveillance programme using material samples for a preliminary investigation of any changes that have taken place in the material properties.

A fluence limitation as demanded by the German regulations is not fulfilled by NPP-T. Other regulations do not contain such a demand for fluence limitation as required in Germany.

To determine the effect of neutron irradiation on the material until the projected end of operations, a special irradiation programme covering end-of-life fluence condition was performed for NPP-T. From the point of view of GRS, the results presented in the POSAR, which also include the results of special investigations of the effects of the neutron embrittlement of the beltline material, indicate that for the reactor pressure vessel, the nil-ductility temperature will remain below 100 °C for the calculated fluence of 40 operating years. However, due to the fact that very different results have been achieved by various organisation with similar materials under sometimes different irradiation conditions, a final assessment of the uncertainties with regard to the expected material changes is currently not possible from the point of view of GRS. Therefore the surveillance programme realised NPP-T has special safety relevance.

The modified arrangement of the suspended samples for the surveillance programme with respect to the location inside the vessel ensures the representativeness of the irradiation conditions and the temperature for the conditions at the reactor pressure vessel wall. A generic deficiency of VVER-1000 plants has thereby been eliminated for NPP-T. The materials chosen for the surveillance samples are representative of the vessel material. The number of samples is sufficient to measure changes in the material with adequate accuracy. The lead factors chosen are small - as is usual in most Western countries - in order to reduce uncertainties due to different flow rates.

- **Assessment of the protection measures against consequential damage in the case of a 2A-break of main-steam lines at the 28.8-m elevation**

In all VVER-1000 plants, the safety systems of the main-steam system (blow-off control and safety valves, main-steam isolating valves) are housed together with the main-steam and feedwater lines in a room outside the containment (28.8-m elevation) without physical separation. As in the case of a pipe failure consequential damage involving adjacent safety-related components or systems may occur which may lead to beyond-design-basis accident sequences, this has been seen as a deficiency (e. g. hazard to the integrity of safety-relevant components) by earlier safety assessments (e. g. by the IAEA), and measures have been recommended.

From the German regulations and German practice it can be concluded that in these cases two different kinds of measures have to be taken as protection against secondary damage due to pipe breaks. One way is to limit the failure of the affected piping to blowdown cross-sections with no further need to assume consequential failures due to pipe whip. The other measure consists of the physical separation of pipes so that pipe failure will remain restricted to the failing pipe itself.

Owing to the design and the overall layout or routing of the piping, conditions at NPP-T do not allow such measures. Therefore the requirements of the current German regulations to be applied here are not fulfilled.

At NPP-T, pipe whip restraints have been installed as partly used in other plants according to international practice. Their design follows the safety requirements of the US regulations, which had to be fulfilled as a requirement of SUJB. Within the framework of the Czech licensing procedure, break locations were identified on the basis of stress criteria and secured by the installation of pipe whip restraints in such a way that breaks in these locations cannot result in consequential damage to other safety-relevant components or systems. The review of GRS has shown that the methodical procedure to determine the postulated break locations and the other parameters (type, position and time of the break) follows the current US regulations, which - in contrast to German standard - are also applied in other Western countries.

Protective measures against consequential damage in the event of postulated leaks or breaks in the main-steam and feedwater lines are assessed by GRS according to the requirements of safety level 3 mainly. GRS was not able to estimate from the available documents how far the measures taken are sufficient to eliminate consequential damage. However, GRS considers the underlying spectrum of break assumptions and the associated consequential damage as too limited and therefore the design solution as not sufficiently robust. In addition, the welding of support plates onto the pressure-retaining wall - although in compliance with the safety standards of different countries - is not seen by GRS to be state-of-the-art for the solution of such problems.

- **Control of small loss-of-coolant accidents (LOCAs) including leaks from the primary into the secondary system**

As the preliminary investigations relating to the standard VVER-1000 have shown, the question of how safely small loss-of-coolant accidents can be controlled has special relevance. Here, steam generator tube failures or a possible limited header lifting involving the challenge of the main-steam relief and safety valves have to be considered in particular. For this type of loss-of-coolant accidents including leaks from the primary to the secondary system, the possible associated release of radioactive substances into the environment has to remain within radiologically admissible limits.

The control of small loss-of-coolant accidents (LOCAs) and leakages flowing not into the containment but into the secondary system was assessed under the aspects listed below. For a classification of the accidents analysed in depth, the list of accidents was reviewed altogether for completeness. The assessments have produced the following results, which are summarised.

– Completeness of the list of accidents

  As regards the scope and systematic structure, there is a considerable improvement compared with the original accident lists for the standard VVER-1000/V-320 plants. The accident list for the NPP-T is closely oriented on the US-NRC guidelines and considers in addition the VVER-1000-specific accident "Steam generator header lifting" which earlier had not been considered as VVER-1000 design basis accident.

  The selection of the design basis accidents - safety level 3 - corresponds to the licensing requirements that are also postulated for modern German plants. The accidents analysed in depth were assessed by GRS in accordance with the requirements of safety level 3. It is current German practice in connection with the Periodic Safety Reviews of German NPPs to analyse as well events that may occur during shutdown states of the reactor plant. These events were not analysed in connection with the safety demonstrations for NPP-T within the framework of the available Safety Analysis Report (Chap. 15) but according to the utility are part of other investigations and especially of the training programme. The scope and methodology of the symptom-based procedures covered in the training programme to control such events were not examined by GRS.

– Quality of the accident analyses

  Parts of the procedure applied by the utility as a basis for the accident analysis were reviewed at random by GRS. The procedure follows US regulations, including the computer codes applied and their validation, the plant data, the conservativeness of the initial and boundary conditions and of the models applied, the sensitivity analyses concerning worst cases, the quality assurance and the documentation. With one exception, they also correspond to the German regulations. This exception is the lacking load combination of the single failure with the maintenance outage, which is only relevant in a few cases in a 3 x 100 % safety system. However, similar exceptions also exist in German licensing practice as well as in other Western countries.

  On the part of the Czech authority and its authorised Technical Safety Organisation, independent conservative audit calculations (accompanied by NRC experts) were performed which confirmed the safety margins indicated in the plant vendor's own calculations. This is in line with German licensing practice and has to be judged positively.

  In connection with the analyses of the leaks from the primary to the secondary system with the targets of preventing overfilling and maintaining reactor core integrity, the adherence to the limits was demonstrated by assuming that in the event of a failure of the main-steam relief valve to open, the main-steam relief valves and the main-steam safety valves-in case of non opening of the affected main-steam relief valves-will fulfil their function even in case of dynamic loading with steam-water mixture. However, there are no plant-specific safety demonstrations in connection with these assumptions.[2] In addition, the case of one of these valves getting inadvertently stuck in open position was not assumed, either. This has either not been justified at all or without demonstrations on a probabilistic level, which is inconsistent with the deterministic requirements for the analysis of design basis accidents.

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[2] see also page 15

– Suitability and reliability of the systems and measures

  As regards the leaks from the primary to the secondary system, considerable improvements have been made which both clearly reduce the frequency of occurrence of these leaks and improve the early detection of indications and actual damage.

  As regards compliance with the protection goals, the assessment concerning the accidents in question is as follows:

  - *Shutdown and preservation of subcriticality:*

    There exist two independent shutdown systems. The shutdown concept is in line with the requirements of the German regulations.

  - *Fuel cooling:*

    As regards the fulfilment of the design-independent requirements of the German regulations for maintaining a sufficient coolant inventory and for residual-heat removal, the investigations of GRS have shown that systems mentioned below differ from the technical solutions applied in Germany:

  – the measures to prevent sump blocking;

  – the isolation systems in the injection lines of the emergency core cooling system.

    The suitability and reliability of these measures and systems have not been analysed by GRS within the framework of the present analyses. Also, as already mentioned in connection with the accident analyses, it has to be demonstrated on plant level that proper functioning of the main-steam relief and safety valves needed to control certain leak accidents is ensured in the event of or after water admission. Otherwise, no deviations from the design-independent requirements of the German regulations were found in connection with the I&C and process-related systems and measures provided for the control of leaks from the primary into the secondary system.

  - *Confinement of radioactive materials*:

    Some individual technical designs for the confinement of radioactive materials differ from German licensing practice. Among them there is no intermediate cooling system between the nuclear area and the service water system. According to the German regulations, this is only dispensable if particularly quality-enhancing and surveillance measures are taken. It was not reviewed by

GRS how far this is the case with the NPP-T.

For the rest, the measures and systems provided are suitable with regard to their design. As concerns the proper functioning of the main-steam relief and safety valves in the event of water admission, the same applies that has been stated above.

Radiological consequences

The radiation exposure levels - calculated in the Safety Analysis Report by Westinghouse according to the procedures prescribed by the NRC - for the considered design basis accidents lie within the NRC acceptance criteria and fulfil the basic licensing requirements of the Czech authority. The formal comparison of the results of the Westinghouse calculations with the accident planning limits according to Section 28 paragraph. 3 shows that in the proximity the 150-mSv limit for the thyroid gland is exceeded. Here, however, it has to be taken into account that the calculation bases of the accident analysis of Westinghouse differ considerably from the German Incident Calculation Bases in their methodology and objectives, which is why the above-mentioned comparison can only serve for orientation. Furthermore, it has to be considered that the method of the Safety Analysis Report to determine the source terms is particularly pessimistic compared with the German regulations. If more realistic assumptions - as the ones used in German licensing practice - are applied, estimates of GRS by engineering judgement, e. g. with regard to the release of iodine, come up with clearly lower radiation exposure levels.

Estimates with the Westinghouse source term show for a receiving point on the German Czech border at 60 km distance from the plant that the values lie clearly below the 150- mSv limit for the thyroid gland and the 50-mSv limit for the effective dose.

- **Ensurance of the capacity of the containment strength in connection with accidents and containment leak rate tests**

In the past, tensile losses have been found in containments for VVER-1000 reactors. When the affected pre-stressing tendons were retightened, some few ones failed completely. To ensure the capacity of the containment strength under accident conditions, measures are required to prevent tensile losses and predict the future development of the pre-stressing forces.

The scope of the assessments was limited to the fulfilment of the requirements in connection with design basis accidents (safety level 3).

– Capacity of the containment strength

  The capacity of the containment strength requires that tensile losses in the pre-stressing tendons or breaks of the pre-stressing tendons are prevented. The building structures to ensure the capacity of the containment strength are in line with the German regulations. For the pre-stressed concrete containment, technological improvements were made to eliminate the known deficiencies of the pre-stressing system, and quality assurance measures were implemented to a sufficient degree. With the installation of additional measuring systems, precautionary measures have been taken which make suitable monitoring of the pre-stressing condition of the containment possible. The capacity of the containment strength is thereby assured.

– Containment leak rate test

  An integrated leak rate test was carried out successfully in accordance with the requirements of the US regulations. The methodical procedure of the integral leak rate test is also comparable to the requirements of the German regulations. The value measured for the leak rate at an overpressure of 400 kPa is very low (0.04 % / 24 h). This has demonstrated a very high degree of leaktightness of the containment in the event of a design basis accident. The low leak rate fulfils the requirements of the German regulations.

- **Protection of the plant in the event of external impacts: aircraft crash and explosions in the proximity of the NPP**

– Aircraft crash

  The frequencies of occurrence of an aircraft crash indicated by the utility for the NPP-T site are clearly lower than those assumed in licensing practice for the design of more recent German NPPs against aircraft crash.

  In the design of NPP-T against the impacts of an aircraft crash, lower load assumptions were laid down than postulated by the German requirements. Analyses in connection with the German reference load case "aircraft crash" - requirements in accordance with safety level 4 - have shown, however, that the containment structures are also able to withstand those loads.

– Explosions

  In the closer surroundings of the plant, there exist no systems or facilities that would jeopardise safety, with the exception of three gas pipelines. On demand of the licensing authority, the applicant has demonstrated with the help of safety analyses that even under pessimistic assumptions of a gas release, there will be no safety-relevant effects. The German requirements for the "Explosion blast wave" load case, which according to the German regulations belong to safety level 4, have also been used as basic requirements in the Czech licensing procedure; the safety demonstrations themselves were not reviewed by GRS.

Summarising assessment

In the view of GRS, the analysis of the seven selected safety issues - which was performed on the basis of the documents and information provided by SUJB and with the restrictions pointed out at the beginning - has shown that the design-independent safety requirements of the German guides and standards are largely fulfilled by the concepts and safety measures provided in NPP-T for the elimination of the known deficiencies of standard VVER-1000 plants. As for the identified deviations, the regulations of other Western countries, in particular the US regulations of the US-NRC, are followed in most cases in line with the demands of the licensing authority and according to the application documents submitted by the applicant.

However, there remain three open points which according to current knowledge and the view derived by GRS are of important safety significance. These are:

– the reliability to function of the main-steam safety and relief valves in case of loading with steam-water mixture

– the safety significance of extended break assumptions for the main-steam and feedwater lines at the 28.8-m elevation and of a resulting even wider spectrum of consequential damage

– the relevance of the discharge rate of the battery capacity especially with regard to the flexibility of accident management measures.

Furthermore, there is a need for clarification in respect of the questions compiled in the Appendix regarding the provision of safety demonstrations.

Part 2

Supplement to the executive summary of the results of the in-depth GRS analysis of selected safety issues relating to the Temelin NPP

- **Taking into account the information provided by SUJB between September 5 and September 21, 2000**

On September 5, 2000, a bilateral German-Czech meeting of BMU/BStMLU and SUJB took place at which the results of the GRS assessment were explained and the three open points and 12 questions requiring clarification were discussed. Before the meeting, SUBJ had already submitted written comments in reply to the open points and the questions requiring clarification. As a result of the meeting it was agreed that SUJB would hand over to GRS additional comments resulting from the technical discussions. The existing answers and comments will be enclosed in the technical part of the GRS report.

On the basis of these assertions and the further written information submitted until September 21, 2000, the statements concerning the three "open points" are supplemented by GRS as follows:

– Reliability to function of the main-steam safety and relief valves in case of loading with steam-water mixture

  SUJB states that the results of the main-steam relief valves tested at the French Cumulus facility also apply for the main-steam relief valves used for the NPP-T. The tested main-steam relief valves show no deviations with regard to their functional mode of operation and the materials used compared with the valves used at NPP-T. Differences only exist in the geometrical dimensions (smaller diameter).
  It was also stated by SUJB that the test results with regard to the safety valves also apply to the main-steam safety valves used at NPP-T since the safety valves are of identical design and made by the same manufacturer. Thus, for SUJB the qualification of the main-steam safety and relief valves with regard to loading with water has been sufficiently demonstrated and clarified in line with the safety requirements.
  The ascertainments regarding this issue transferred by SUJB are plausible to GRS. However, this GRS assessment is not based on a comprehension of the validations

and evaluations representing the basis for the SUJB decisions. The meaningfulness of the GRS assessment is in so far limited.

- Safety significance of extended break assumptions for the main-steam and feedwater lines at the 28.8-m elevation and of a resulting even wider spectrum of consequential damage

  This point remains open also in consideration of the more recent information provided.

- Relevance of the discharge rate of the battery capacity especially with regard to the flexibility of accident management measures

  From the documents submitted it becomes clear that the actual capacity in the real plant is more than 120 minutes and that further power supplies can be realised at short notice.
  Taking this into account, this open point has been resolved.

As for the 12 questions for which GRS saw a need for clarification, the situation is as follows:

In connection with questions 1, 2, 3, 4, 6 and 10, SUJB provided comprehensible information which from the point of view of GRS confirms that the underlying technical issues have been clarified.

In connection with questions 5,7,8,9,11 and 12, the authority did provide further-reaching information to clarify these technical questions which demonstrate that from the point of view of the authority they have been answered. However, GRS cannot confirm that there has been a final clarification as this would require an additional in-depth review.

Appendix

Compilation of questions by GRS requiring clarification with regard to kind and scope of safety demonstrations in connection with the in-depth assessments of the selected safety issues relating to the Temelin NPP

In the document " Executive summary of the in-depth GRS analysis of selected safety issues relating to the Temelin NPP", the deviations from the German regulations in connection with seven safety issues that were assessed in depth are described. The assessment of these deviations showed that - apart from three open points that are addressed in the executive summary - there is also a need for clarification with regard to the kind and scope of safety demonstrations. The answers to these questions provided by SUJB are contained in the technical part of the GRS report.

- **Effects of the modified core design**

Question 1:
How is the suitability of the measures to monitor the drop times of the control rods and fuel rod deformation demonstrated?

Question 2:
How is it ensured that the examination of the applicability of the calculation codes used for the core design by recalculation of results from the commissioning phase covers all aspects that are essential for a validation?

- **Interfaces with the digital safety I&C system**

Question 3:
Which technical and administrative measures (e. g. introduction of repair time limits) ensure the safety functions also in those cases where the failure combination "random failure and maintenance outage" has not been considered for the reactor protection system?

Question 4:
In the area of data acquisition and diesel load logic (non-programmable logic) of the reactor protection system, how is the lack of equipment-related diversity compensated, and which demonstrations exist in this respect?

- **Safe control of small loss-of-coolant accidents (LOCAs) and leaks from the primary to the secondary system**

Question 5:

How is the required high degree of reliability to function of the non-isolable main-steam relief valves and main-steam safety valves demonstrated with regard to their closing after previous opening?

Question 6:

How is it ensured that sufficient demonstrations exist concerning the safe control of representative accidents in connection with shutdown plant states?

Question 7:

Is it ensured that independent audit calculations - also regarding leaks from the primary into the secondary system - are performed and that in this context sufficient coolant supplies are also demonstrated for the long-term period (> 1h)?

Question 8:

How is it achieved that the measures to ensure a sufficient coolant backflow into the sump in the event of a LOCA will be equal to the method proposed by the OECD in 1999 and 2000?

Question 9:

How is it demonstrated that the valves in the injection lines of the emergency core cooling system, which are in "closed" position during normal operation, will perform their opening function with a high degree of reliability even under accident conditions?

Question 10:

How is it demonstrated for the component cooling system of the reactor coolant pumps with regard to the securing and monitoring of interfaces between high-pressure and low-pressure systems that leaks in the heat exchanger to the primary system are safely controlled?

Question 11:

How is the lacking intermediate cooling system between the nuclear area and the secured service water system compensated by quality-enhancing and surveillance measures?

Question 12:

Which further-reaching examinations are planned to clarify the causes of the different results of the dose calculations that were performed by the Westinghouse and Skoda companies for the cases "steam generator tube rupture" and "steam generator header lifting"?