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ETE Road Map

According to Chapter IV and V of the "Conclusions of the Melk Process and Follow-Up"

Item 7b
Severe Accidents Related Issues

Preliminary Monitoring Report

Report to the Federal Ministry of Agriculture, Forestry, Environment and Water Management of Austria

Vienna, August 2004

ENCONET Consulting

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EXECUTIVE SUMMARY

I. Basis and the background for the project

The Republic of Austria and the Czech Republic have, using the good offices of Commissioner Verheugen, reached an accord on the “Conclusions of the Melk Process and Follow-up” on 29 November 2001. In order to enable an effective use of the “Melk Process” achievements in the area of nuclear safety, the Annex I of this “Brussels Agreement” contains details on specific actions to be taken as a follow-up to the “trialogue“ of the “Melk Process” in the framework of the pertinent Czech-Austrian Bilateral Agreement.

Furthermore, the Commission on the Assessment of Environmental Impact of the Temelín NPP - set up based on a resolution of the Government of the Czech Republic - presented a report and recommended in its Position the implementation of twenty-one concrete measures (Annex II of the “Brussels Agreement”).

The signatories agreed that the implementation of the said measures would also be regularly monitored jointly by Czech and Austrian experts within the Czech-Austrian Bilateral Agreement.

A “Roadmap” regarding the monitoring on the technical level in the framework of the pertinent Czech-Austrian Bilateral Agreement as foreseen in the “Brussels Agreement” has been elaborated and agreed by the Deputy Prime Minister and the Minister of Foreign Affairs of the Czech Republic and the Minister of Agriculture and Forestry, Environment and Water Management of the Republic of Austria on 10 December 2001.

The Federal Ministry of Agriculture, Forestry, Environment and Water Management entrusted the Umweltbundesamt (Federal Environment Agency) with the general management of the implementation of the “Roadmap”. Each entry to the “Roadmap” corresponds to a specific technical project [see ANNEX C].

The objective of the Roadmap process covered by the item 7 as stated in Annex I of the “Brussels Agreement” is: *"Effective prevention and mitigation of consequences of beyond design basis accidents (severe accidents)”*.

Annex I provides the following statements regarding the *“Present Status and specific Actions Planned”:*

“A set of preventive and mitigative measures is, at present, applied in NPP Temelín with respect to beyond design basis accidents. These include software and hardware measures, among others, e.g. Symptom Based Emergency Operating Procedures, Technical Support Centre, Post Accident Monitoring System, Emergency Preparedness.

For the purpose of emergency preparedness, the PSA was employed with the aim to identify and group events with different initiating occurrences, but with similar end-effects. On the basis of this assessment the relative risk was estimated for specific events in order to select those, which will serve for the determination of emergency response activities (pre-planned, reactive).

Severe Accidents Management Guidelines (SAMG) as a state-of the-art tool will complete the whole system of mitigation measures with respect to the beyond design basis accident management. The project for SAMG development is scheduled to be finished by end 2002 to be followed by validation.

To foster mutual understanding two lines of activities will be followed within the framework of the bilateral agreement:

a) A Working Group on comparison of calculations regarding the radiological consequences of BDBA with a view to harmonise the basis for emergency preparedness will be established.

b) The exchange of information related to SAMG will include discussion on the analytical basis as well as on corresponding software and hardware measures. "

The issue (a) has been covered in a separate project PN1 [see ANNEX C], the issue (b) is covered by this project.

Refering to Chapter IV of the "Brussels Agreement" and the principles of the "Roadmap", a number of issues identified in the "trialogue" of the Melk Process are found suitable to be followed-up in the framework of the Bilateral Agreement. The following seven issues are closely related to the topic of item No. 7 (b) and are therefore also covered in this project:

- Issue No. 1 Containment bypass and preliminary-to-secondary (PRISE) leakage accidents
- Issue No. 4 Containment Design and Arrangement
- Issue No. 5 Probabilistic Safety Assessment and Severe Accidents
- Issue No. 6 Emergency Operating Procedures EOPs & Severe Accident Management Guidelines (SAMGs)
- Issue No. 16 Hydrogen Control
- Issue No. 26 Beyond Design Bases Accident Analysis
- Issue No. 29 Technical Basis for Temelín Emergency Planning Zones (EPZs)

The Roadmap specified that the related Specialists' Workshop would be held in the 1st half of 2003 to discuss this issue.This workshop on the "Roadmap" item No. 7 was conducted in Prague on 17 and 18 June 2003 according to Article 7 (4) of the Bilateral Agreement of the Exchange of Information on Nuclear Safety. This workshop was the key element in the monitoring process. The analysis of information made available there played a significant role in the development of the basis for the Preliminary Monitoring Report.

A Specialists' Team of international experts was committed by the Umweltbundesamt (Federal Environment Agency) on behalf of the Austrian Government to provide technical support for the monitoring on the technical level of the implementation of the SAMGs Issue as listed in Annex I of the Conclusions of the Melk Process and Follow-up. This specific technical project is referred to as project PN7 comprising altogether seven predefined "project milestones" (PMs).

Consideration of beyond design basis accidents (BDBAs) of the NPP is an essential component of the defence in depth approach used in nuclear safety. BDBAs have low likelihood, but may have significant consequences resulting from the degradation of nuclear fuel. It is worth noting that accidents of low likelihood, but more severe than those taken into account in the design basis (i.e. BDBAs), have not been explicitly considered in the design of nuclear power plants which are currently under operation. However, the possibility of severe accidents has been recognized later as an important safety issue and addressed in comprehensive and systematic way for most of the operating plants.

In accordance with the current safety philosophy the consideration of severe accidents in NPPs usually includes the following elements:

- Identification of event sequences that lead to severe accidents;
- Consideration of existing plant capabilities, including the possible use of some systems beyond their originally intended function, to return the plant to a controlled state and to mitigate the consequences of the severe accident;
- Identification of the permissible degree of non-uniformities in the hydrogen distribution in the atmosphere,
- Evaluation of potential design changes which could either reduce the likelihood of these events or mitigate the consequences;
- Establishing accident management procedures, based on representative and dominant severe accidents.

The set of actions taken during the evolution of an event sequence towards a design basis accident (DBA) is known as Emergency Management and once the sequence enters any BDBA sequence – Accident Management (AM). The AM is intended to prevent the escalation of the event into a severe accident, to mitigate the consequences of a severe accident, to re-establish critical safety functions and to return the plant to a controllable safe state. Accident Management Programmes (AMP) based on this concept were adopted in many nuclear plants starting from early 1980s. AMPs comprise plans and actions undertaken to ensure that personnel with responsibilities for AM are adequately prepared to take effective on-site actions to prevent or to mitigate the consequences of a severe accident and, when deemed necessary, to plan and implement plant functional modifications.

The Temelín Nuclear Power Plant (NPP), being of an original Soviet design, which was later upgraded with equipment following western philosophy, addressed severe accident management (SAM) in the later construction phase. Considering that the process of SAM implementation is not completed and information on the approach taken at the Temelín NPP is insufficient the “Severe accident related issues” remained as one of the items to be addressed during the follow up to the Melk process.

II. The objectives of the PN7 project

Enconet and IRR-ARCS defined the following additional objectives in order to implement this overall objective:

- The **principal objective** of the project is to provide technical support and the expertise to assess the adequacy of the development and implementation of SAMG at the Temelín NPP, as well as the overall approach to the issue of severe accidents at Temelín. (see also the list of issues to be tackled according to the “Brussels agreement” (ANNEX I).
- The **specific objective** of the **horizontal segment** is to assure that all issues that need to be addressed within evaluation of Severe Accident Management for Temelín are highlighted to allow for an evaluation considering state-of-the-art requirements and criteria as applied in European Union (EU) and/or the United States (USA).
- The **specific objective** of the **vertical segment** is to assure that specific contributors to early containment failure (such as hydrogen combustion) are included within the evaluation of SAM for Temelín.

III. Main tasks accomplished within PN7 project

III.1 Identification of severe accidents issues and scenarios

The issues, which are of relevance for understanding the phenomenology of severe accidents, have been identified and relevant information collected. This includes all the accident phenomena, which have been identified (and for which specific reactors were analyzed) in the western countries and for Soviet designed reactors. Also issues, which are being discussed in research fora in Europe and in the USA, have been added to the list.

After the issues had been identified, their applicability for a WWER 1000 was investigated. This work resulted in a list of issues, which are of relevance for a WWER 1000 and Temelín in particular. This issue list has served as a basis for the further investigation of Severe Accident issues important for Temelín.

III.2 Identification of approaches to prevention and mitigation of severe accidents

In addition to the identification of scenarios, the issue of prevention of severe accidents and approaches to mitigation have been investigated, using the western practice as well as the findings of the severe accident analyses of WWER 1000 units.

III.3 Review of applicability of Severe Accident issues and prevention/mitigation to Temelín

The specific applicability of severe accident issues to Temelín has been investigated to evaluate, which of the issues are of relevance and of interest for further studies. For this purpose information was drawn from other WWERs 1000 plants and considering Temelín specific upgrades which are of relevance.

From the list of issues, which are applicable to Temelín, those, which are open issues or potentially problem issues in Temelín have been selected.

The review of regulatory approaches included the position of US Nuclear Regulatory Commission (US NRC) on PWRs with large dry containments of US design, of licensing authorities within the EU such as the French IRSN, the German Reaktorsicherheits-kommission (RSK) and of Western European technical support organizations (TSO) such as French-German consortium GRS/Riskaudit.

The US NRC for their ruling has determined that dry containments offer such large safety margins that for the existing US plants which implement WOG Westinghouse SAMGs, the hydrogen combustion and direct containment heating issues can be considered resolved.

Licensing authorities within the EU apply various approaches in their respective countries. Detailed findings are presented in Annex A to this report.

The analysis of the actual situation in Temelín NPP showed that the plant is provided with a large dry containment, with comparable features to those in use with US pressurised water reactors (PWRs), but differing in some geometrical aspects, in particular in the shape of steam generator boxes, which are horizontal and not vertical as in the PWRs. This can result in slower dispersal or propagation of hydrogen released during not only the in-vessel accident phase and in possible increase of its local concentration to the values higher than in typical PWR containments. On the other hand, compared to the US plants, Temelín has the advantage of the installed hydrogen recombiners, which deplete hydrogen by combining it with oxygen to water over long-term operation. Detailed findings are presented in the report and summarized in Section 4.3.

III.4 Code calculations for assessment of selected sequences

The initial activity in this step was identification and selection of accident sequences leading to an immediate thread to the integrity of the containment either due to Core Concrete Interaction failure or due to Hydrogen formation leading to a detonable gas mixture within the containment eventually resulting in detonation pressure bulid up. Both these cases were analysed using MELCOR, a computer code suitable to analyse to the detail required severe accidents and determine consequences. The MELCOR analyses have been done within both the horizontal segment and the vertical segment of PN7. The WWER 1000 MELCOR input deck available for Kozloduy NPP (KNPP) plant has been modified taking into consideration specifics of Temelín insofar as they were known. The results have been assessed to verify which of the problem issues need to be specifically addressed for the development of SAMGs.

The calculations made in PN7 covered more than twelve scenarios with some variants aimed at checking sensitivity of results to the assumptions adopted in calculations. Several points in which no sufficient data were available to judge Temelín statements have been identified, but generally the agreement of PN7 calculations results with those of TACIS programme for WWER 1000 NPPs and with Czech results for Temelín was reasonable.

As it was recognized that besides general review there is a need for in-depth analysis of the topics connected with hydrogen hazards, the problems of hydrogen generation and transient local distributions were addressed in MELCOR analyses of three scenarios plus a special 3-dimensional GASFLOW analysis for a specific scenario, providing insights into non-uniformities of hydrogen distribution during the phase of most intensive hydrogen release and the related hazards.

III.5 Identification of relevant steps in the development of SAMGs

With consideration of the severe accident sequences, their outcomes, probabilities, and possible mitigation measures, the steps relevant to the development of SAMGs have been established, including specifics to be addressed in Severe Accident Management Guidelines (SAMGs).

Based on experience of the project team in evaluation and validation of SAMGs, the specifics to be investigated in this area have been identified.

The issues have been listed, which need to be addressed in relation to the adaptation of SAMGs and training of plant staff in the use of SAMGs.

The Severe Accident Management requirements have been analysed as indicated by international practice and formulated in the Westinghouse Owners Group (WOG) SAMGs. The available information on the Temelín approach and the SAMG development status was reviewed. The result showed that the approach followed by Temelín corresponds to good international practice and after completion - expected by the end of 2004 - the SAMGs in Temelín NPP should be equivalent to those in other plants using the WOG approach. Some minor points needing further monitoring have been found as identified below.

III.6 Verifiable line items

The objective of this task was to break down the overall subject into the line items, which could then be verified for completeness and compliance with the accepted international practice. This task was the “road map” for the whole project. On the basis of all the analyses, which are discussed above, the project team has identified all necessary elements, which are of interest in developing and implementing an acceptable severe accident management program including its verification. The list of Verifiable Line Items (VLIs) covering more than 240 questions in 40 topical areas has been developed, covering both SAMG development and accident sequences in the plant. It was the basis for consolidation of the information achieved during the joint workshop with representatives of the Safety Authority and the Temelín NPP operator, which took place in June 2003 in Prague.

III.7 Specialists Workshop

The preparatory activities for the workshop included the development of briefing material and a briefing session for the Austrian delegation, proposing experts to participate in the workshop as well as participation in the workshop. The compliance and differences with the state of the art practices have been identified and commented on for their safety significance.

A list of documents was prepared, the Specific Information Request, that considered to contain that kind of information required to provide profound answers to the VLIs.

In the Workshop the Czech side presented a set of twelve papers, which together with the discussion sessions made it possible to determine the answers to most VLIs, as shown in the report. Some methodological aspects have been left unanswered due to the limitations of time available and the complex nature of the phenomena involved in severe accidents analysis. Nevertheless, the information accumulated in the preparation of the workshop and obtained during the workshop is sufficient to formulate a coarse picture of the Temelín NPPs preparation to cope with severe accidents, (given the fact that Sections 3.1.4, 3.1.5, 3.2.3, 3.2.4, 3.3.1, 3.3.2, 3.3.3, 3.3.4, 3.3.5, 3.3.6, 3.4.3, 3.6.2, and 3.6.5 of the main report discuss areas where the information presented was evaluated as insufficient; in addition, see Section 1.4 of the main report for an explanation of the assessment framework). In some cases the Austrian delegation requested further written information and the Czech side provided it soon after the workshop. This information was subsequently used to repeat a selection of the calculations performed within PN7 project with updated characteristics of basemat concrete and hydrogen recombiners in the Temelín NPP. The final conclusions in the report are based on those updated calculations.

IV. Main findings

IV.1 Regulatory approach and practice

The Czech Nuclear Regulatory Authority (SUJB) has required the plant to prepare and accomplish a program to deal with BDBAs, including estimation of plant vulnerabilities, proposed accident management procedures and the schedule of their implementation. The targets set by SUJB for severe core damage frequency and for large off-site releases are to underscore 10^{-4} and 10^{-5} per reactor year, respectively, which is consistent with the INSAG targets for existing NPPs.

The responsibility for development of SAMGs is left to the utility. The regulatory body defines acceptance criteria and provides guidance to Temelín NPP, leaving enough flexibility for potential candidate actions to address specific challenges.

IV.2 Temelín programme of severe accident management

The development and implementation of Temelín SAM programme has not been finalized, however, it is well advanced.

The overall concept and approach to development/implementation of SAMG package was found to reflect the current good practice in the SAM area. The selection of plant specific SAM strategies has been based on the well-established generic approach developed by Westinghouse Owners Group. These generic strategies have been adapted to Temelín plant conditions based on a systematic process that reflects the current state-of-the-art in this area.

The programme is supported by severe accident analysis and plant specific probabilistic safety assessment (PSA). However, there were some instances when the existing results of SA analysis were not properly incorporated into the PSA. It should be noted that also some SAM strategies, apparently the most recent, are not well supported by severe accident (SA) analysis. The interface between the PSA team and thermal hydraulic analysis team needs improvement.

The calculation tools used for SA analysis are similar to those used worldwide for the purpose of SAM and the team that has been responsible for calculations is qualified. The existing analyses provide a reasonable basis for understanding plant specific vulnerabilities to severe accidents and the identification of AM strategies. Some of the existing analyses are old and do not necessarily reflect the current plant status and state-of-the-art in the area of SA codes, modelling and simulation, in particular with respect to hydrogen distribution within the containment and with respect to the discharge of molten core material in case of a pressure vessel defect. The plant is planning to improve these analyses using more current codes and improved modelling concepts.

The PSA study includes Level 1 and 2. The first version of PSA has been reviewed during an IAEA mission and the resulting recommendations are reported to be incorporated into the upgraded study. However, the upgraded PSA is still not finalized. Generally, the 1996 PSA study was developed in compliance with the current state-of-the-art, and the updated analysis was intended to address IAEA comments and the as-built design of the plant. The PN7 team has observed some deficiencies, but they are not expected to have significant impact on the final conclusions with regard to SAM strategies. The existing results have been used in the development of SAMG strategies and setting up priorities in the execution of strategies.

Westinghouse in close co-operation with plant staff has developed a plant specific SAMG package. The contents, structure, and format of plant specific SAMG, which were shown at the Workshop, have been found to reflect the current state-of-the-art practice. This package is currently under internal review and translation into Czech language.

Organizational arrangements related to SAMG have not been finalized yet. Although the upgraded ERP Emergency Response Plan has been submitted to SUJB for approval, the updated version of the Emergency Operating Procedures including transition points to SAMGs need to be developed and implemented. Some concerns can be raised in the definition of responsibilities/authorities for determination and approval of an intentional release of radioactive material during a SA. The staffing of SAMG Evaluation Group within the Technical Support Centre is another issue that is not clear enough..It is recommended that the Austrian Government addresses these aspects in the ongoing joint monitoring process on technical level.

The plant has properly considered all further steps of SAMG implementation including validation and training and plans for their execution are being developed. Based on the available knowledge all the related plant arrangements are considered adequate. Little is known also about the training and refreshing courses of SAM staff and the related schedules for implementation. Therefore, it would be welcome if the related activities would be subjected to further joint monitoring in the framework of the pertinent bilateral Agreement between Austria and the Czech Republic. It should be noted that proper evaluation of the SAMG package including the supporting analyses would require detailed investigations that involve specialized expertise and considerable effort. Such an evaluation was beyond the scope of PN 7 project. Therefore, it would be very desirable to have detailed aspects of SAM development and implementation addressed by qualified independent external reviewers. It is known that the plant management and SUJB seriously consider having an independent review of SAM (i.e. a IAEA RAMP mission).

IV.3 Technical measures available in Temelín for SA management.

One of the main areas of hazards due to severe accidents is that of primary to secondary circuit leakages, since such leakages involve loss of coolant accidents with the leak point situated outside the containment. In case of such an accident all four barriers preventing radioactivity release to the environment can be lost simultaneously. Both contemporary regulatory guidance and industrial practice stress the necessity to avoid large PRISE events. In Temelín the hazards involved in primary to secondary leakage (PRISE) accidents are well recognized, the appropriate strategies developed and the technical means are provided to cope with PRISE events.

Another potential hazard is connected with long term complete loss of electric power, both from outside sources and from emergency diesel generators installed at the NPP (station blackout). In such a case the means of heat removal from the reactor are lost, except for gradual evaporation of water, first in the secondary, then in the primary coolant circuit. If this situation persists for several hours, the coolant in the core evaporates, the core dries out, and will be damaged.

The preventive measures at Temelín NPP correctly address the issue of station blackout. The most important measure for mitigation of the effects of blackout and other transients involving loss of electric power consists in forced depressurization of the primary circuit. While not comprehensive, summary presentations of calculations carried out by the Czech experts as well as calculations performed within the PN7 project indicate that the capability for depressurization in Temelín is comparable with that in other plants of similar vintage and is sufficient for timely depressurization of RCS. The Temelín NPP has two lines of defence in this respect (the primary relief valve, PORV, and the emergency gas removal system, EGRS). The WOG SAM strategies being implemented in the plant recognize the importance of depressurization and the EGRS, although of limited capacity, can serve as an additional means of depressurization in the unlikely case of a severe accident with PORV failure. Moreover, the measures taken to prevent a blackout seem to be satisfactory.

In view of the long delays of core damage in case of blackout, the limited capacity of batteries in Temelín seems to be inappropriate. According to the design the period of time that the batteries are sufficient for plant control is shorter than the time that would pass before severe damage of the core. Thus the potential advantages of good thermal hydraulic properties of Temelín could not be used due to battery limitations. Temelín EOPs and SAM strategies include measures to extend battery power supply time by re-structuring the load profile much beyond the design period of 1 hour. Nevertheless, it would be desirable to exchange batteries or include into the system additional power sources providing electric power during station blackout.

An important safety advantage of Temelín NPP is the fact that it is provided with a large dry containment. This reduces considerably the challenges to containment integrity during severe accidents. Similarly as in other NPPs with large dry containment, the hazards of early containment failure due to direct containment heating (DCH) in Temelín NPP have been evaluated as negligible and the strategy of reactor coolant system (RCS) depressurization included in SAM in Temelín further reduces such hazards. The long-term pressurization hazards are reduced by the fact that the basemat concrete in place in Temelín practically does not contain any carbon, so there is no buildup of carbon monoxide and carbon dioxide due to molten corium-concrete interaction. This reduces the long term quantities of noncondensible gases inside the containment. The calculations with the MELCOR code showed that the containment integrity is not threatened by long term increases of pressure due to gas generation. Rather, the calculations show that basemat failure occurs long before overpressure failure becomes an issue.

Hydrogen hazards in NPPs with large dry containment are considered to be unimportant by US NRC and some regulatory bodies in EU countries, but most EU regulatory bodies require technical means for hydrogen depletion. In Temelín the release rates of hydrogen during the in-vessel phase of the accident are comparable with those in PWRs, and the volume of the containment is similar. The geometry of the steam generator boxes and the ducts in Temelion NPP is different from that in PWRs and makes hydrogen mixing less effective, which in case of small break loss-of-coolant accident (SB LOCA) can lead to local formation of sensitive clouds of hydrogen during the in-vessel accident phase. The mean frequency of the accident scenario is about $1{,}7\times10^{-7}$ per year (given the accident sequence frequency, a high likelihood of ignition, and a 50% chance of a detonation given ignition; the assumption is conservatively made that the detonation leads directly to containment failure with a large source term). Even with an arbitrarily large source term, the mean consequence would be of the order of 50 000 person-Sv (calculated over a year's worth of weather conditions). The product of the mean severe accident frequency and the mean consequence would approximately represent the overall risk to the public per year of operation (note that this is a conservative calculation which assumes a 50% chance of a detonation leading to a large containment leak):

$$(1{,}7\times10^{-7}\ 1\ [1/a]) \times (50\ 000\ \text{person-Sv}) = 8{,}5\times10^{-3}\ \text{person-Sv/a}$$

In the ex-vessel phase the presence of a large dry containment and early inerting of the containment by steam contribute to prevention of hydrogen hazards. In the long term the installed hydrogen recombination system designed for DBA conditions, but passively operating also under severe conditions, will contribute to containment inerting by reducing the hydrogen and oxygen content. However, this process is slow and for severe accidents it would be advantageous to have properly located PARs of higher capacity.

The Czech strategy consists of:

a) early intentional hydrogen deflagration (through planned actuation of equipment to try to initiate a deflagration), which should help reduce formation of sensitive clouds during in-vessel phase;
b) reliance on the hydrogen recombiners (PARs) to gradually reduce the hydrogen source in the containment;
c) long term inerting of containment with steam during the ex-vessel phase with procedural controls on sprary actuation to prevent de-inerting burns; and
d) as necesary, venting the containment through a high pressure venting line through filters to the plant stack to release hydrogen from the containment.

Both Czech and PN7 calculations showed that in the case of unplanned actuation of the containment spray system at the moment when the contents of hydrogen is the highest the containment integrity could be lost, and Czech materials provide an evaluation of radiological consequences of such a scenario. However, the SAM strategy proposed for Temelín addresses the issue of reduction of the hazards of late confinement failure due to hydrogen deflagration in line with the Westinghouse SAMG approach. In the case of ultimate necessity, Temelín can actuate as an option the containment pressure test filtered venting system to reduce containment pressure or hydrogen content. This issue seems to be still under development. As the heating due to fission product collection in filters can result in rising filter temperatures (with loss of filter efficiency) or in the worst case induce filter burning, the issues of filtered venting in Temelín should be further monitored.

The main severe accident hazard consists in the possibility of containment basemat penetration.

The measures planned to be implemented in Temelín in case of RPV failure at low pressure assure slowing down of the molten corium concrete interaction (MCCI) process. While these measures go in the right direction, it cannot be proved that they assure protection of the basemat against penetration by molten corium if RPV failure occurs. The likelihood of reactor pressure vessel (RPV) failure is small, as shown by recent analysis, but it exists. According to the statements of Czech specialists, the measures planned in Temelín include corium spreading and water-cooling, which together with the planned remote opening of the cavity door should enable to stop the corium progression.

The calculations performed within PN7 project confirmed that corium spreading slows down the process and provides additional time margins. The effectiveness of water-cooling was not studied in PN7 due to the lack of access to the latest experimental OECD data. Recent information about the results of large scale tests on concrete penetration by molten corium conducted within OECD programme on "The Melt Coolability and Concrete Interaction" indicates that in large scale test in the US enhanced cooling was obtained due to long term water cooling of the molten corium mass. Other experimental studies in Germany in this matter indicate some limitations for the reduction of the core melt attack by top cooling of released core melt with water. The Czech Republic participates actively in some programmes and has the actual information available on the OECD MCCI program.

As of now, the stopping of the corium erosion progress cannot be clearly demonstrated. Therefore, the Temelín staff considers additional measures aimed at improving leaktightness of rooms below the containment basemat. The hazards due to radioactive releases in case of basemat melt-through are much smaller than in the case of an early containment rupture. As shown elsewhere, for releases due to basemat failure, the mass of radioactive aerosols still suspended in the containment atmosphere is dramatically reduced (orders of magnitude) compared with early containment failure. Not considering re-volatilisation and emanation of deposited contaminants during late containment failure, the offsite radiological hazards are correspondingly reduced.

During the Prague meeting, in response to questions the Czech specialists discussed the environment in the reactor building after melt-through of the containment basemat. The Czech experts discussed an evolving strategy of attempting to prevent re-volatilization of fission products that have already been deposited on surfaces in the containment; this could result from violent air turbulence in case the containment would depressurize when the basemat melts through. The evolving strategy also includes prevention of hydrogen combustion in the reactor building after basemat melt-through. The reason for this is to preserve reactor building integrity to allow both for natural aerosol attenuation mechanisms to lower the source term, and to allow the release of fractions of the gas content from the reactor building via the plant stack (via the reactor building ventilation system) to achieve greater dispersion and lower radiation doses offsite.

The strategy would involve depressurization of the containment – before basemat melt-through – via the venting system (high pressure duct work to the plant stack). This would also reduce the hydrogen concentration in the containment. During the Prague meeting Czech specialists mentioned these issues, but no detailed information was obtained on the approach being followed.

The measures and strategies to reduce fission product releases are in keeping with the international practice. The open issues are mostly connected with the reduction of radiological releases in the case of basemat penetration by molten corium. Czech specialists consider it a problem for future consideration, while they see as the most urgent tasks those, which are related to prevention of the basemat melt-through.

V. Recommendations for Further Monitoring

The monitoring process conducted so far within the framework of the “Brussels agreement” (Annex I) in the area of severe accidents helped to clarify a number of relevant issues. It was demonstrated that a comprehensive process directed towards accomplishing the comprehensive SAM and mitigation of SA consequences is in place at Temelín NPP. However, this process is still ongoing and the Specialist’s Team at the present can only follow a number of views and expectations on the SAMs final implementation as expressed by the Czech side.

The Specialist’s Team would recommend the Austrian Governement the consideration of revisiting the findings in the framework of the pertinent bilateral Agreement between Austria and the Czech Republic.

The following areas were identified as of interest:

- The supporting severe accident analysis and PSA as well as their use in the verification of SAM strategies and the related procedures,
- SAMG implementation activities including procedural framework, SAMG validation, and SAM related staff training,
- Identification of the permissible degree of non-uniformities in the hydrogen distribution in the atmosphere
- Implementation of plant changes to enhance the technical measures for SAM.

More detailed discussion of the proposed monitoring issues in these areas is provided below.

The Specialist’s Team would recommend the Austrian Governement the consideration of revisiting the calculations to be made by Temelín NPP using MELCOR 1.8.5 and other code systems, and consider to obtain more detailed and verified information on:

- The capabilities of PORV, together with the effectiveness of the planned coolant system depressurization procedure,
- The regulatory framework for and effectiveness of hydrogen control, and/or additional use of filtered venting for mitigation of radioactive releases,
- Operational capabilities of the emergency gas removal system in SA conditions,
- Analyses of basemat meltthrough failure.

The Specialist’s Team would also recommend the Austrian Governement the consideration of revisiting the SAMG implementation activities at Temelín in order to confirm that the remaining steps of the implementation process are successfully completed. Important items that need further monitoring/verification include the revised procedural framework, SAMG validation, and staff training process. At the same time the recommendations from any independent review of SAM and their resolution should be paid due attention by the Austrian Government.

Technical measures needed for prevention and mitigation of risk significant scenarios should be monitored to demonstrate that appropriate plant arrangements are in place (both procedures and hardware measures). Due attention should be given to SA situations that are most relevant from safety point of view such as basemat penetration in case of molten corium release from the RPV and station blackout. Aspects, which are worth to be mentioned in this context include the measures for timely opening of the reactor cavity door before the RPV failure, protection of containment penetrations and the containment liner against MCCI, and increasing the capacity of batteries. Further analytical work conducted by the plant and the TSO staff on the MCCI hazards and mitigation of the related radiological consequences should also be monitored.

ZUSAMMENFASSUNG

I. Grundlage und Hintergrund des Projektes

Die Republik Österreich und die Tschechische Republik haben mit Unterstützung des Kommissars Verheugen am 29. November 2001 Übereinstimmung hinsichtlich der "Schlussfolgerungen des Melker Prozesses und das Follow-up" erzielt. Um eine wirksame Umsetzung der Ergebnisse des Melker Prozesses im Bereich der nuklearen Sicherheit zu ermöglichen, enthält der Anhang I dieses „Brüsseler Abkommens" Details zu spezifischen Maßnahmen, die als Follow-up zum „Trialog" des „Melker Prozesses" im Rahmen des betreffenden bilateralen tschechisch-österreichischen Abkommens durchzuführen sind.

Weiters legte die Kommission zur Prüfung der Umweltverträglichkeit des KKWs Temelin, die auf Grund einer Resolution der Regierung der Tschechischen Republik eingesetzt wurde, einen Bericht vor und schlug in ihrer Stellungnahme die Umsetzung einundzwanzig konkreter Maßnahmen vor (Anhang II des „Brüsseler Abkommens").

Die Unterzeichner kamen überein, die Umsetzung der genannten Maßnahmen gemeinsam von tschechischen und österreichischen Experten im Rahmen des bilateralen Abkommens über den Austausch von Informationen regelmäßig zu überwachen.

Zur Überwachung auf technischer Ebene im Rahmen des einschlägigen tschechisch-österreichischen bilateralen Abkommens wurde, wie im „Brüsseler Abkommen" vorgesehen, eine „Roadmap" („Fahrplan") ausgearbeitet und am 10. Dezember 2001 vom stellvertretenden Premierminister und Außenminister der Tschechischen Republik, sowie vom Bundesminister für Land- und Forstwirtschaft, Umwelt und Wasserwirtschaft der Republik Österreich, vereinbart.

Das österreichische Bundesministerium für Land- und Forstwirtschaft, Umwelt und Wasserwirtschaft beauftragte das Umweltbundesamt mit der Gesamtkoordination der Umsetzung der „Roadmap". Jeder Eintrag in der „Roadmap" entspricht einem spezifischen technischen Projekt [siehe ANNEX C].

Die Zielsetzung von Punkt 7 des „Roadmap" Prozesses, wie in Annex I des „Brüsseler Abkommens" festgehalten, lautet: *„Wirksame Vermeidung und Eingrenzung von Folgewirkungen aus Auslegungsüberschreitenden Unfällen (Schweren Unfällen)"*.

In Annex I werden folgende Aussagen bezüglich dem *„Derzeitigen Stand und den geplanten spezifischen Maßnahmen"* festgehalten :

„Hinsichtlich auslegungsüberschreitender Störfälle gibt es im KKW Temelin derzeit eine Reihe von Maßnahmen, die die Folgen von Störfällen vermeiden oder verringern sollen. Diese schließen unter anderem Software - und Hardwaremaßnahmen, wie zum Beispiel symptomorientierte Notfallbetriebsvorschriften, technisches Hilfszentrum, Überwachungssystem für Störfallfolgen, Notfallvorsorge mit ein. Zum Zwecke der Notfallvorsorge wurde die PSA (Probabilistic Safety Assessment) angewendet, mit dem Ziel, Ereignisse mit verschiedenen Eintrittshäufigkeiten, aber ähnlichen Endeffekten, zu identifizieren und in Gruppen einzuteilen. Auf der Grundlage dieser Beurteilung wurde das relative Risiko für spezifische Ereignisse eingeschätzt, um jene davon auszuwählen, die zur Bestimmung von Notfallmaßnahmen dienen werden (vorausgeplante, reaktive). Richtlinien zum Management schwerer Unfälle (SAMG) als ein dem letzten Stand entsprechendes Instrument werden das gesamte System der folgenverringernden Maßnahmen bezüglich des Managements von auslegungsüberschreitenden Störfällen ergänzen.

Der Abschluss des Projekts für die Entwicklung der SAMG ist für Ende 2002 geplant und wird anschließend evaluiert werden. Zur Förderung des beiderseitigen Verständnisses werden im Rahmen des bilateralen Abkommens Tätigkeiten in zwei Richtungen hin unternommen:

a) Eine Arbeitsgruppe zum Vergleich von Berechnungen hinsichtlich der radiologischen Folgen von BDBA (Beyond Design Base Accident) wird eingerichtet, um die Grundlagen für eine Notfallvorsorge zu harmonisieren.

b) Der Informationsaustausch zu den SAMG wird die Erörterung der analytischen Grundlage wie auch der entsprechenden Software – und Hardwaremaßnahmen mit einschließen.“

Punkt (a) wird in einem anderen Projekt (PN1) [siehe Annex C] abgehandelt. Punkt (b) wird vom gegenständlichen Projekt (PN7) abgedeckt.

Bezugnehmend auf Kapitel IV des "Brüsseler Abkommens" und auf die Prinzipien der "Roadmap" werden eine Reihe von Problemkreisen, die im "Trialog" des "Melker Prozesses" erarbeitet wurden, als verfolgungswürdig im Rahmen des bilateralen Abkommens erachtet. Die folgenden sieben Problemkreise stehen in enger Verbindung mit Punkt 7 (b) und werden aus diesem Grund ebenfalls in diesem Projekt behandelt:

- Problemkreis 1 Sicherheitsbehälter-Bypass und Störfälle mit Leckagen vom Primär- in den Sekundärkühlkreislauf (PRISE)
- Problemkreis 4 Sicherheitsbehälter-Auslegung und Anordnung
- Problemkreis 5 Probabilistische Sicherheitsanalyse und Schwere Unfälle
- Problemkreis 6 Störfallbetriebsanleitungen (EOPs) & Richtlinien zur Beherrschung von Schweren Unfällen (SAMGs)
- Problemkreis 16 Wasserstoff-Einstellung
- Problemkreis 26 Analyse von Auslegungsüberschreitenden Unfällen
- Problemkreis 29 Technische Grundlage für die Gültigkeitszonen des Alarmplans für Temelín (EPZs)

Die „Roadmap“ sah für die erste Hälfte des Jahres 2003 einen Experten-Workshop zur Erörterung dieser Thematik vor. Dieser Experten-Workshop zum Punkt 7b der “Road Map” wurde am 17. und 18. Juni 2003, gemäß Artikel 7 (4) des bilateralen Abkommens über den Austausch von Informationen über Nukleare Sicherheit in Prag abgehalten. Dieses Treffen war das Schlüsselereignis im Monitoringprozess. Die Auswertung der damals zur Verfügung gestellten Informationen spielte eine entscheidende Rolle für die Entwicklung der Grundlagen für den Vorläufigen Monitoringbericht.

Ein internationales Experten-Team wurde vom Umweltbundesamt im Namen der österreichischen Regierung mit dem technischen Support zur Überwachung der Implementierung der SAMG-Thematik auf technischer Ebene, wie im Anhang I der „Schlussfolgerungen des Melker Prozesses und des Follow-up“ aufgezeigt, beauftragt. Dieses technische Projekt wird als "PN7-Projekt" bezeichnet, welches insgesamt sieben vorgegebene „Projektmeilensteine“ (PM) umfasst.

Die Überlegungen zu Auslegungsüberschreitenden Unfällen (BDBAs) von Kernkraftwerken sind ein wesentlicher Bestandteil des im Bereich nukleare Sicherheit angewandten "Defense-in-Depth" Konzepts. Auslegungsüberschreitende Unfälle haben geringe Eintrittswahrscheinlichkeiten, können jedoch gravierende Folgewirkungen nach sich ziehen, die aus dem Versagen des Kernbrennstoffs resultieren. Es ist zu beachten, dass Unfälle die zwar eine geringe Eintrittswahrscheinlichkeit haben, jedoch schwerer sind, als die Auslegungsstörfälle jener Kernkraftwerke, die derzeit in Betrieb sind (d.h. auslegungsüberschreitende Unfälle) bei deren Auslegung nicht ausdrücklich in Betracht gezogen wurden. Die Möglichkeit von Schweren Unfällen wurde später als wichtige Sicherheitsfrage erkannt und für die meisten Anlagen in umfassender und systematischer Weise angesprochen.

In Übereinstimmung mit der gegenwärtigen Sicherheitsphilosophie schließt die Berücksichtigung von schweren Unfällen üblicherweise folgende Elemente mit ein:

- Ermittlung von Ereignisabfolgen, die zu schweren Unfällen führen;
- Berücksichtigung der vorhandenen Leistungsfähigkeit der Anlagen, einschließlich des möglichen Einsatzes einiger Systeme über die ursprünglich beabsichtigten Funktionsweisen hinaus, um die Anlage in einen beherrschbaren Zustand überzuführen und die Folgewirkungen eines schweren Unfalles zu begrenzen;
- Ermittlung der zulässigen Ungleichverteilung von Wasserstoff im Freiraum der Sicherheitshülle;
- Auswertungen von Möglichkeiten zu Auslegungsabänderungen, welche die Eintrittswahrscheinlichkeit solcher Ereignisse vermindern oder deren Folgen einschränken könnten;
- Erstellen von Störfallbeherrschungsmaßnahmen auf der Grundlage von repräsentativen und dominierenden Schweren Unfällen.

Der Katalog von Maßnahmen, die getroffen werden, wenn eine Ereignisabfolge sich in Richtung eines Auslegungsstörfalles (DBA) entwickelt, ist unter der Bezeichnung "Störfall Management" bekannt, und - sobald der Ablauf einer der Auslegungsüberschreitenden Unfallsequenzen (BDBA) folgt - als "Unfallmanagement" (Accident Management - AM). Das Unfallmanagement soll die Eskalation des Ereignisses zu einem Schweren Unfall vermeiden, die Folgen eines Schweren Unfalls verringern, die kritischen Sicherheitsfunktionen wiederherstellen und die Anlage in einen sicheren, überwachbaren Zustand überführen. Auf diesen Grundzügen aufbauende Unfallmanagement-Programme (AMP) wurden in zahlreichen kerntechnischen Anlagen - beginnend in den frühen 80er Jahren - eingeführt. Unfallmanagement-Programme umfassen Pläne und Maßnahmen, um sicherzustellen, dass die Angestellten, die für das Unfallmanagement verantwortlich sind, ausreichend darauf vorbereitet sind, wirksame Maßnahmen vor Ort zu treffen, um die Folgen eines schweren Unfalls zu vermeiden oder abzumildern und, sofern es als notwendig angesehen wird, Änderungen in der Funktionsweise der Anlage zu planen und einzurichten.

Im Kernkraftwerk Temelín (NPP) – einem KKW sowjetischer Bauart, welches später mit Ausrüstungselementen, die der westlichen Philosophie folgen, nachgerüstet wurde – hat man das Management von schweren Unfällen in der späten Errichtungsphase angesprochen. Da der Vorgang zur Implementierung des Management von Schweren Unfällen noch nicht abgeschlossen ist und die Informationen über die Vorgangsweise im KKW Temelín nicht ausreichend sind, ist der Punkt "Schwere Unfälle“ einer jener Bereiche, die über den „Melker Prozesses“ hinaus weiter zu behandeln sein werden.

II. Die Zielsetzungen des PN7 Projektes

Enconet und IRR-ARCS haben folgende zusätzliche Ziele definiert, die zur Realisierung des Hauptzieles führen sollen:

- Das **Hauptziel** des Projekts ist es, technische Unterstützung und Sachkenntnis bereitzustellen, um die Angemessenheit der Entwicklung und Einführung von SAMGs im KKW Temelín zu beurteilen, ebenso wie den allgemeinen Umgang mit dem Thema "Schwere Unfälle im KKW Temelín" (siehe auch die Liste der laut „Brüsseler Abkommen“ (ANNEX I) zu behandelnden Fragen).

- Das **spezifische Ziel** des **horizontalen Segments** ist es, sicherzustellen, dass alle Themen angesprochen werden, die im Rahmen der Bewertung des Managements von schweren Unfällen bearbeitet werden müssen, um eine Auswertung unter Bedachtnahme auf die Anforderungen des Standes der Technik, sowie der in der Europäischen Union (EU) und/ oder den Vereinigten Staaten von Amerika (USA) angewandten Kriterien zu ermöglichen.
- Das **spezifische Ziel** des **vertikalen Segments** ist es, sicherzustellen, dass Vorgänge, die zum frühzeitigen Versagen der Sicherheitshülle beitragen (wie das Verbrennen von Wasserstoffgas) in die Auswertung der SAM für Temelín einbezogen werden.

III. Hauptaufgabenstellungen, die im Rahmen des PN7-Projektes bearbeitet wurden

III.1 Ermittlung von Sachverhalten und Szenarien für Schwere Störfälle

Jene Sachverhalte, die für das Verständnis der Phänomenologie Schwerer Unfälle wesentlich sind, wurden ermittelt und wichtige Informationen gesammelt. Dies umfasst alle Unfallphänomene, die in den westlichen Staaten und für Reaktoren sowjetischer Bauart ermittelt worden sind (und die für einzelne Reaktoren untersucht worden sind). Auch Themen, die in den Forschungsforen in Europa und in den USA laufend diskutiert werden, wurden in die Liste aufgenommen.

Nach der Ermittlung der Sachverhalte wurde untersucht, ob sie auf einen WWER 1000 anwendbar sind. Das Ergebnis dieser Arbeit war eine Liste von Sachverhalten, die für einen WWER 1000 und Temelín im Besonderen von Bedeutung sind. Diese Liste diente als Grundlage für die weiteren Untersuchungen von schweren Unfällen mit Relevanz für Temelín.

III.2 Ermittlung von Ansätzen zur Vermeidung und Eingrenzung von schweren Unfällen

Zusätzlich zur Ermittlung von Szenarien ist der Themenkreis "Vermeidung von schweren Unfällen und Ansätze zu deren Einschränkung" untersucht worden. Zu diesem Zweck wurden sowohl die westlichen Gepflogenheiten herangezogen, als auch die Erkenntnisse aus den Analysen Schwerer Unfälle von WWER 1000 Blöcken.

III.3 Überprüfung der Relevanz von Problemkreisen auf dem Gebiet "Schwere Unfälle" und deren Verhinderung/Eingrenzung für Temelín

Die spezifische Relevanz für Temelín von Problemkreisen bei Schweren Unfällen wurde untersucht, um auszuwerten, welche der Sachverhalte für die weiteren Untersuchungen wesentlich und daher von Interesse sind. Zu diesem Zweck wurden Daten von anderen WWER 1000 Anlagen herangezogen und Temelín spezifische Verbesserungen berücksichtigt.

Aus der Liste der für Temelín relevanten Problemkreise wurden diejenigen ausgewählt, die offene Fragen oder mögliche Problempunkte darstellen.

Die Durchsicht der Vorgangsweisen der Aufsichtsbehörden schloss die Position der US Nuclear Regulatory Commission (US-NRC) zu den Druckwasserreaktoren mit großen trockenen Sicherheitsbehältern US-amerikanischer Bauart ein, sowie die Positionen von Genehmigungsbehörden in der EU, wie der französischen IRSN, der deutschen Reaktorsicherheitskommission (RSK) und jene von westeuropäischen Technischen Support Firmen (TSO), wie dem deutsch-französischen Konsortium GRS/Riskaudit.

Die US-NRC hat für ihren Verantwortungsbereich festgestellt, dass große trockene Sicherheitsbehälter derart umfangreiche Sicherheitsreserven bereithalten, dass für die in Betrieb befindlichen US Anlagen, die WOG Westinghouse SAMGs einsetzen, die Wasserstoffverbrennung und das direkte Aufheizen des Sicherheitsbehälters als gelöst betrachtet werden können.

Die Vorgehensweise der Genehmigungsbehörden innerhalb der EU ist in den jeweiligen Mitgliedsländern unterschiedlich. Detaillierte Angaben dazu werden in ANNEX A zu diesem Bericht vorgestellt.

Die Analyse der im KKW Temelín gegebenen Situation hat gezeigt, dass die Anlage mit einem großen trockenen Sicherheitsbehälter ausgestattet ist, dessen Einrichtungen mit denen von US Druckwasserreaktoren (PWRs) vergleichbar sind, der aber in etlichen geometrischen Aspekten abweicht, im Speziellen bei der Form der Dampferzeugerboxen, die horizontal ausgeführt sind und nicht vertikal wie bei den PWRs. Dies kann zur langsameren Verdünnung oder Ausbreitung von Wasserstoff nicht nur in der Freisetzungsphase innerhalb des Druckgefäßes führen, und damit zu einem potentiellen Anstieg der lokalen Konzentrationen auf Werte, die über jenen in typischen Sicherheitsbehältern von PWRs liegen. Auf der anderen Seite hat Temelín - verglichen mit US Anlagen - den Vorteil, katalytische Wasserstoffrekombinatoren installiert zu haben, die Wasserstoff abreichern, indem sie ihn im Langzeitbetrieb mit Sauerstoff zu Wasser umwandeln. Detaillierte Erkenntnisse werden im Bericht vorgestellt und im Abschnitt 4.3 zusammengefasst.

III.4 Computerberechnungen zur Bewertung ausgewählter Störfallabläufe

Die Eingangstätigkeit für diesen Schritt war das Ausfindigmachen und die Auswahl von Störfallabläufen, welche zu einer unmittelbaren Gefährdung der Integrität des Sicherheitsbehälters führen, entweder durch Versagen als Folge der Kernschmelzereaktion mit dem Sicherheitsbehälterbeton oder durch Wasserstoffbildung, die schließlich zu einem detonationsfähigen Gasgemisch im Sicherheitsbehälter führen kann und zu einem Druckaufbau als Folge der Detonation. Diese beiden Fälle werden mit Hilfe von MELCOR untersucht, einem Programm zur detaillierten Analyse von schweren Unfällen und zur Feststellung der Folgewirkungen. Die MELCOR Untersuchungen wurden sowohl im horizontalen als auch im vertikalen Segment des PN7 durchgeführt. Die WWER 1000 MELCOR-Eingabedaten, die für das KKW Kozloduy (KNPP) zur Verfügung standen, wurden unter Bedachtnahme der Besonderheiten von Temelín, soweit bekannt, abgeändert. Die Ergebnisse wurden beurteilt, um klarzustellen, auf welche Problemstellungen bei der Entwicklung der SAMGs besonderes Augenmerk gelegt werden muss.

Die Berechnungen im Rahmen von PN7 umfassten mehr als 12 Szenarien, wobei einige Varianten das Ziel hatten, die Ergebnisse auf deren Sensitivität hinsichtlich der Annahmen, auf denen die Berechnungen basierten, zu prüfen. Es wurde eine Reihe von Punkten gefunden, für die nicht ausreichend Daten zur Verfügung standen, um die Aussagen zu Temelin zu beurteilen. Im Allgemeinen war die Übereinstimmung der Ergebnisse der PN7-Berechnungen mit denen des TACIS Programmes für WWER 1000 KKWs und mit den tschechischen Ergebnissen zu Temelín angemessen.

Da festgestellt wurde, dass neben der generellen Beurteilung eine detaillierte Analyse der Fragen betreffend die Wasserstoffgefährdung erforderlich ist, wurden die Probleme der Wasserstoff-Produktion und der lokalen Wasserstoffverteilungen während der Transienten in den MELCOR Analysen in drei Szenarien behandelt. Zusätzlich wurde eine spezielle dreidimensionale GASFLOW-Analyse für ein spezifisches Szenario durchgeführt, das Einsichten in unausgeglichene Wasserstoffverteilungen während der Phase der intensivsten Wasserstofffreisetzungen und in die damit verbundenen Gefahrenmomente bot.

III.5 Ermittlung der wesentlichen Schritte bei der Entwicklung von SAMGs

Aus der Betrachtung der Abläufe von schweren Unfällen, deren Folgewirkungen, Wahrscheinlichkeiten und möglichen Begrenzungsmaßnahmen wurden jene Schritte festgelegt, die für die Entwicklung von SAMGs wesentlich sind, einschließlich der Besonderheiten, die in den SAMGs behandelt werden müssen.

Auf der Erfahrung der Projektgruppe in Bezug auf die Bewertung und Anwendbarkeitsüberprüfung von SAMGs aufbauend, wurden die in diesem Bereich zu untersuchenden Details identifiziert.

Es wurden jene Problemkreise aufgelistet, die hinsichtlich der Anpassung der SAMGs und der Schulung der Betriebsmannschaft betreffend die Verwendung der SAMGs behandelt werden müssen.

Die Anforderungen betreffend das Management Schwerer Unfälle sind internationalen Gepflogenheiten und den Westinghouse Owners Group (WOG) SAMGs entsprechend untersucht worden. Die verfügbaren Informationen über den Ansatz in Temelín und den Entwicklungsstand der SAMGs wurden analysiert. Das Ergebnis hat gezeigt, dass der Ansatz, der in Temelín verfolgt wird, mit der üblichen internationalen Praxis übereinstimmt und dass nach Vervollständigung – geplant für Ende 2004 – die SAMGs im KKW Temelín gleichwertig mit jenen anderen Anlagen sein sollten, die auf dem WOG Ansatz aufbauen. Einige kleinere Punkte - wie weiter unten angeführt - wurden gefunden, die weitere Beobachtung erforderlich machen.

III.6 Verifiable Line Items (VLIs)

Ziel dieser Tätigkeit war es, die Gesamtfragestellung in Einzelaspekte zu unterteilen, die dann auf Vollständigkeit und Übereinstimmung mit der anerkannten internationalen Praxis geprüft werden konnten. Diese Überprüfungstätigkeit führte zum Ablaufplan für das Gesamtprojekt. Auf der Grundlage aller oben angeführten Untersuchungen hat das Projektteam die Elemente ermittelt, die für die Entwicklung und die Einführung eines annehmbaren Programms für das Management von schweren Unfällen, einschließlich dessen Verifikation, notwendig sind. Eine Liste von Verifiable Line Items (VLIs), die mehr als 240 Fragen aus 40 Themenbereichen enthält, wurde erstellt. Sie deckt sowohl die SAMG Entwicklung als auch die Unfallabläufe in der Anlage ab. Diese Liste war die Grundlage für die Konsolidierung der Informationen, die während des im Juni 2003 in Prag abgehaltenen Workshops von den Vertretern der Aufsichtsbehörde und der Betreiber des KKW Temelín zur Verfügung gestellt wurden.

III.7 Expertentreffen (Workshop)

Die vorbereitenden Tätigkeiten zum Expertentreffen umfassten die Zusammenstellung des Informationsmaterials und eine Informationsveranstaltung für die österreichische Delegation, die Nennung von Experten, die am Workshop teilnehmen sollten, sowie die Teilnahme am Workshop selbst. Übereinstimmungen und Abweichungen vom Stand der Technik wurden festgestellt und im Hinblick auf deren Bedeutung für die Sicherheit kommentiert.

Eine Liste jener Dokumente, wurde erstellt, die jene Informationen enthalten, die zur profunden Beantwortung der VLIs erforderlich wären („Specific Information Request“).

Die von tschechischer Seite beim Expertentreffen präsentierten zwölf Vorträge, zusammen mit der anschließenden Diskussion, ermöglichten es, auf die meisten der VLIs Antworten zu geben (siehe Bericht). Einige Aspekte zur Methodologie sind auf Grund der Zeitbeschränkungen und der Komplexität, der mit der Analyse Schwerer Unfälle verbundenen Phänomene unbeantwortet geblieben. Nichtsdestoweniger sind die für die Vorbereitung des Expertentreffens gesammelten und die beim Expertentreffen erhaltenen Informationen ausreichend, ein grobes Bild der Vorbereitungen des KKW Temelín zur Bewältigung von schweren Unfällen zu vermitteln (dabei ist zu beachten, dass die Abschnitte 3.1.4, 3.1.5, 3.2.3, 3.2.4, 3.3.1, 3.3.2, 3.3.3, 3.3.4, 3.3.5, 3.3.6, 3.4.3, 3.6.2,und 3.6.5 des Hauptteils des Berichts Sachbereiche behandeln, in denen die gelieferte Information als unzureichend bewertet wurde; zusätzlich ist die in Abschnitt 1.4 des Hauptteils des Berichts enthaltene Erklärung der Beurteilungsrandbedingungen zu berücksichtigen). In einigen Fällen hat die österreichische Delegation weiterreichende schriftliche Informationen gefordert, die die tschechische Seite kurz nach dem Expertentreffen geliefert hat. Diese Information wurde in der Folge verwendet, um eine Auswahl der Berechnungen im PN7-Projekt mit aktualisierten Eigenschaften für den Beton der Bodenplatte und die Wasserstoffrekombinatoren zu wiederholen. Die endgültigen Schlüsse im Bericht basieren auf diesen verbesserten Berechnungen.

IV Hauptergebnisse

IV.1 Genehmigungsansatz und -praxis

Die tschechische Aufsichtsbehörde (SUJB) hat die Anlagenbetreiber dazu aufgefordert, ein Programm zur Bewältigung Auslegungsüberschreitender Unfälle vorzubereiten und zu implementieren, einschließlich einer Einschätzung der Verwundbarkeit der Anlage, eines Programmes zum Management von schweren Unfällen, sowie eines Zeitplanes für dessen Einführung. Die Zielvorgabe, welche von SUJB für die Häufigkeit schwerer Schäden am Reaktorkern und für Auslegungsüberschreitende Freisetzungen in die Umgebung festgelegt wurde, lautet, dass 10^{-4} bzw. 10^{-5} Ereignisse pro Betriebsjahr unterschritten werden müssen. Diese Werte entsprechen den INSAG Zielvorgaben für in Betrieb befindliche KKWs .

Die Verantwortung für die Entwicklung der SAMGs bleibt den Anlagenbetreibern überlassen. Die Aufsichtsbehörde schreibt die Annahmekriterien vor und gibt Richtungshinweise an das KKW Temelín, die ausreichenden Spielraum für mögliche Maßnahmen lassen, um spezielle Herausforderungen zu bewältigen.

IV.2 Das Temelín Programm für das Management von schweren Unfällen

Die Entwicklung und Implementierung des Programms für das Management von schweren Unfällen in Temelin wurde zwar noch nicht abgeschlossen, ist jedoch weit fortgeschritten.

Das generelle Konzept und der Ansatz für die Entwicklung/Implementierung des SAMG-Paketes entspricht der gegenwärtig geübten Praxis im Bereich SAM Die Auswahl von anlagenspezifischen SAM Strategien fand auf der Grundlage des bewährten, von der Westinghouse Owners Group entwickelten Ansatzes statt. Diese generischen Strategien wurden, basierend auf einem systematischen Prozess, der ebenfalls dem gegenwärtigen Stand der Technik auf diesem Gebiet entspricht, an die Anlagebedingungen in Temelín angepasst.

Das Programm stützt sich auf Analysen der schweren Unfälle und auf eine anlagenspezifische Probabilistische Sicherheitsanalyse (PSA). Es gab jedoch einige Fälle, in denen die vorliegenden Ergebnisse der Analysen Schwerer Unfälle nicht in geeigneter Form in die Probabilistische Sicherheitsanalyse übertragen wurden. Es muss festgestellt werden, dass auch einige der SAM Strategien – offensichtlich die zuletzt eingebrachten – nicht ausreichend mittels Analysen Schwerer Unfälle (SA) abgesichert sind. Die Schnittstelle zwischen der PSA-Gruppe und der Thermohydraulik-Analyse-Gruppe sollte verbessert werden.

Die Berechnungshilfsmittel, die für SA-Analysen verwendet werden, ähneln jenen, die weltweit für SAM-Zwecke eingesetzt werden. Die Gruppe, die für diese Berechnungen verantwortlich ist, ist entsprechend qualifiziert. Die vorhandenen Analysen bieten eine vernünftige Grundlage zum Verständnis der Gefährdung der Anlage durch Schwere Unfälle und zur Ermittlung von Unfallmanagement-Strategien. Einige der vorhandenen Analysen sind älteren Datums und spiegeln weder unbedingt den aktuellen Anlagenzustand wider noch sind sie auf dem neuesten Stand der Technik auf dem Gebiet der Analyse schwerer Unfälle, deren Modellierung und Simulation, insbesondere in Hinsicht auf die Wasserstoffverteilung innerhalb des Sicherheitsbehälters und auf das Ausstoßen von Kernschmelzematerial im Fall eines Reaktordruckbehälterdefekts. Die Betreiber der Anlage planen, diese Analysen durch die Verwendung neuerer Rechenprogramme und durch verbesserte Modellierungskonzepte zu verbessern.

Die PSA Studie umfasst Level 1 und 2. Die Erstversion der PSA wurde im Zuge einer IAEA Mission überprüft und es wurde gesagt, dass die daraus resultierenden Empfehlungen in eine aktualisierte Studie aufgenommen werden sollen. Allerdings wurde die aktualisierte PSA noch immer nicht fertig gestellt. Die PSA 1996 wurde entsprechend dem [damaligen] Stand der Technik entwickelt; die aktualisierte PSA soll den Kommentaren der IAEA Rechnung tragen, sowie auch dem „as built“ Anlagenzustand. Das PN7-Experten-Team hat einige Mängel festgestellt, von denen aber nicht zu erwarten ist, dass sie einen wesentlichen Einfluss auf die Schlussfolgerungen hinsichtlich der SAM Strategien haben werden. Die vorliegenden Ergebnisse wurden für die Entwicklung von SAMG Strategien und das Erstellen von Prioritäten für die Ausführung der Strategien herangezogen.

Westinghouse hat in enger Zusammenarbeit mit dem Betreiberpersonal ein anlagenspezifisches SAMG Paket entwickelt. Inhalt, Struktur und Format von anlagenspezifischen SAMGs, wie sie beim Expertentreffen gezeigt wurden, wurden als dem Stand der Technik entsprechend eingestuft. Das Paket wird derzeit intern überprüft und ins Tschechische übersetzt.

Organisatorische Vorkehrungen für SAMG wurden noch nicht fertig gestellt. Obwohl der verbesserte ERP (Emergency Response Plan = Örtlicher und Überörtlicher Alarmplan) der SUJB zur Genehmigung vorgelegt wurde, muss die verbesserte Version der Störfallmaßnahmen (EOPs) einschließlich der Schnittstellen zu den SAMGs noch entwickelt und implementiert werden. Eine Reihe von Einwänden können zu den Festlegungen von Verantwortlichkeiten/ Entscheidungsträgern zur Feststellung und Genehmigung beabsichtigter Freisetzung von radioaktivem Material während eines schweren Unfalles vorgebracht werden. Die Bestellung der Mitglieder der SAMG Auswertungs-Gruppe im Technischen Support Zentrum ist ein weiterer Punkt, der nicht klar genug definiert ist. Es wird empfohlen, dass die Österreichische Regierung diese Aspekte im laufenden gemeinsamen Beobachtungsprozess auf technischer Ebene anspricht.

Die Anlagenbetreiber haben alle weiteren Schritte zur Einführung der SAMG ordnungsgemäß berücksichtigt, einschließlich der Anwendbarkeitsüberprüfung, der Schulung und den Durchführungsplänen. Auf der Grundlage des vorhandenen Wissens werden die entsprechenden Dispositionen als adäquat betrachtet. Zur Schulung und zu den Auffrischungskursen für das SAM Personal und zu den diesbezüglichen Zeitplänen liegen nur wenige Informationen vor. Aus diesem Grund wäre es wünschenswert, die entsprechenden Aktivitäten zum Gegenstand einer gemeinsamen Beobachtung im Rahmen des gegenwärtigen bilateralen Abkommens zwischen Österreich und der Tschechischen Republik zu machen. Es ist festzuhalten, dass eine tiefgreifende Auswertung des SAMG Paketes einschließlich der begleitenden Analysen detaillierte Untersuchungen, sehr spezielles Sachwissen und enormen Aufwand erfordern würde, die den Rahmen dieses PN 7-Projektes gesprengt hätten. Es wäre allerdings höchst wünschenswert, Einzelheiten der SAM Entwicklung und der Implementierung von qualifizierten, unabhängigen, externen Personen auswerten zu lassen. Bekanntlich überlegen sowohl das Management der Anlage und das SUJB ernsthaft eine unabhängige Überprüfung des SAM (d. h. eine IAEA-RAMP Mission).

IV.3 In Temelín verfügbare technische Maßnahmen für das SA Management

Einer der Hauptbereiche für Gefährdungen, die von schweren Unfällen ausgehen, kommt von Leckagen des Primärkühlkreislaufs in den Sekundärkühlkreislauf (PRISE), da derartige Leckagen einen Kühlmittelverluststörfall mit einer Austrittsstelle außerhalb des Sicherheitsbehälters bedeuten. Kommt es zu einem derartigen Unfall, können alle vier Barrieren, die sonst eine Freisetzung von radioaktivem Material in die Umwelt verhindern, gleichzeitig versagen. Aktuelle Richtlinien der Aufsichtsbehörden, wie auch die geübte Industriepraxis, unterstreichen die Notwendigkeit einer Vermeidung von großen Leckagen des Primärkühlkreislaufs in den Sekundärkühlkreislauf (PRISE). In Temelín hat man die Gefahrenmomente, die Unfälle mit Leckagen des Primärkühlkreislaufs in den Sekundärkühlkreislauf mit sich bringen, sehr wohl erkannt und entsprechende Strategien entwickelt und technische Vorkehrungen getroffen, um mit PRISE Ereignissen umgehen zu können.

Ein weiteres Gefährdungsmoment ist mit dem längerfristigen, vollständigen Verlust der Stromversorgung verbunden, sowohl was die Versorgung von außen betrifft, als auch jene von den Notstromdieseln, die im KKW installiert sind, (station blackout). In einem solchen Fall gehen alle Möglichkeiten der Wärmeabfuhr aus dem Reaktor verloren, mit Ausnahme des stetigen Verdampfens des Wassers, zuerst im Sekundär-, dann im Primärkreislauf. Wenn diese Lage für mehrere Stunden anhält, verdampft das Kühlmittel im Reaktorkern, der Kern trocknet aus und wird beschädigt.

Die Vorbeugungsmaßnahmen in Temelín sind richtigerweise darauf ausgerichtet, das Problem eines Totalausfalles der Stromversorgung zu verhindern. Die wichtigste Maßnahme zur Begrenzung der Auswirkungen eines Totalausfalls der Stromversorgung und von anderen Transienten, die mit einem Stromversorgungsausfall einhergehen, ist die vorgezogene, erzwungene Druckentlastung des Primärkreislaufs. Obgleich nicht umfassend, lassen die zusammenfassenden Darstellungen der Berechnungen der tschechischen Experten während des Workshops, sowie auch die Berechnungen im Rahmen des PN7-Projektes - erkennen, dass die Leistungsfähigkeit für die Druckentlastung in Temelín mit der von Anlagen aus derselben Errichtungsperiode vergleichbar ist und für eine zeitgerechte Druckentlastung des Reaktorkühlsystems (RCS) ausreichend ist. Das KKW Temelin hat diesbezüglich zwei Verteidigungslinien: die Primären Entlastungsventile (PORV), und das Gasabfuhr-Notstandssystem (EGRS). Die WOG SAM Strategien tragen in der Form, in der sie in der Anlage verwendet werden, der Wichtigkeit der Druckentlastung Rechnung, und das EGRS, selbst wenn dieses nur eine begrenzte Leistungsfähigkeit besitzt, kann doch als zusätzliche Entlastungseinrichtung im unwahrscheinlichen Szenario eines schweren Unfalles mit Versagen der PORV funktionieren. Darüber hinaus erscheinen die Maßnahmen zur Verhinderung eines Totalversagens der Stromversorgung zufriedenstellend.

In Anbetracht des langen Hinauszögerns von Kernschädigungen bei totalem Stromausfall, erscheint die begrenzte Versorgungsdauer durch die Batterien in Temelín unangemessen. Entsprechend der Auslegung ist der Zeitraum, in dem die Batterien für die Anlagenregelung ausreichend sind, kürzer, als der Zeitraum, der verstreichen würde, bevor der Reaktorkern schwer beschädigt werden würde. Daher kommen die durch gute thermo-hydraulische Eigenschaften erzielten Vorteile von Temelín aufgrund der begrenzten Batteriekapazität nicht zum Tragen. Die EOPs und SAM Strategien von Temelín sehen auch Maßnahmen zur Ausweitung der Versorgungsdauer mit Batteriestrom durch Restrukturierung des Belastungsprofiles auf weit über eine Stunde vor. Dessen ungeachtet wäre es wünschenswert, die Batterien auszutauschen, oder das Gesamtsystem mit weiteren Stromquellen zu verbinden, die während eines Totalausfalls der Stromversorgung elektrischen Strom liefern.

Ein wichtiger Sicherheitsvorteil des KKW Temelín ist die Tatsache, dass es mit einem großen, trockenen Sicherheitsbehälter ausgestattet ist. Diese Tatsache vermindert die Anforderungen an die Integrität des Sicherheitsbehälters während Schwerer Unfälle beträchtlich. Ähnlich wie in anderen KKWs mit großen, trockenen Sicherheitsbehältern, wurde die Gefährdung durch frühes Sicherheitsbehälterversagen auf Grund von direkter Aufheizung der Sicherheitsbehälterwand (DCH) für Temelín als vernachlässigbar eingestuft und die Strategie der Reaktorkühlsystem-Druckentlastung, welche in den SAM in Temelín einbezogen wurde, vermindert solche Gefährdungen weiter. Die Gefährdung durch Langzeitdruckaufbau wird dadurch gemindert, dass der Beton der Tragplatte praktisch keinen Kohlenstoff enthält und sich damit kein Kohlenmonoxid und -dioxid durch die Einwirkung von geschmolzenem Corium auf den Beton aufbauen kann. Dadurch werden die Mengen von nichtkondensierbaren Gasen weniger, die sich über lange Zeit im Sicherheitsbehälter bilden. Die Berechnungen mit dem MELCOR Programm haben gezeigt, dass die Integrität des Sicherheitsbehälters durch den langzeitigen Druckaufbau durch Gasproduktion nicht gefährdet ist. Die Berechnungen weisen eher darauf hin, dass das Versagen der Bodenplatte schon lange vor dem Überdruckversagen ein Problem wird.

Wasserstoffgefährdung in KKWs mit großen, trockenen Sicherheitsbehältern werden von der US-NRC und von einigen EU-Staaten als unwichtig eingestuft. Die meisten EU-Aufsichtsbehörden fordern jedoch technische Einrichtungen zur Verminderung des Wasserstoffgehaltes. In Temelín sind die Freisetzungsraten für Wasserstoff während der Phase des Unfalls im Reaktordruckbehälter vergleichbar mit jenen in Druckwasserreaktoren vergleichbarer westlicher Bautypen, und auch das freie Sicherheitsbehältervolumen weist eine ähnliche Größenordnung auf. Die geometrische Anordnung der Dampferzeugerboxen und der Überströmkanäle im KKW Temelín unterscheidet sich von den Druckwasserreaktoren vergleichbarer westlicher Bauart und macht die Durchmischung von Wasserstoff weniger effizient, was für den Fall eines Kühlmittelverluststörfalles mit kleinem Leck (SB LOCA) zu einer lokalen Freisetzung und Bildung von sensitiven Wolken von Wasserstoff während der Unfallphase im Reaktordruckbehälter führen kann. Die mittlere Frequenz für das Unfallszenario ist $1{,}7\ .10^{-7}$ Ereignisse pro Reaktorbetriebsjahr (bei gegebener Frequenz für diese Unfallsequenz, einer hohen Entzündungswahrscheinlichkeit und einem 50%igen Wahrscheinlichkeit, dass daraus eine Detonation bei Zündung wird; aus Konservativitätsgründen wurde die Annahme getroffen, dass die Detonation direkt zum Versagen der Sicherheitshülle mit einem hohen Quellterm führt). Selbst mit einem sehr hoch angesetzen Quellterm läge die gemittelte Folgewirkung im Bereich von 50 000 Mann-Sv (errechnet über ein Jahr anhand von realen Wetterbedingungen). Das Produkt aus der mittleren Frequenz für den Schweren Unfall und den mittleren Auswirkungen würde ungefähr das mittlere Gesamtrisiko [Anm. aus dem jeweiligen Unfall] für die Bevölkerung während eines Reaktorbetriebsjahres darstellen. (Hier ist zu bemerken, dass dies eine konservative Annahme ist, indem es ein 50%ige Wahrscheinlichkeit für eine Detonation annimmt, die [Anm. zwingend] zu einem großen Sicherheitsbehälterleck führt):

$$(1{,}7\times10^{-7}\ 1\ [1/a])\times(50\ 000\ \text{Mann-Sv}) = 8{,}5\times10^{-3}\ \text{Mann-Sv/a}$$

In der Unfallphase, die außerhalb des Reaktordruckbehälters stattfindet, ist das Vorhandensein eines großen trockenen Sicherheitsbehälters und einer frühzeitigen Inertisierung des Sicherheitsbehälters durch Dampf ein Beitrag zur Verhinderung von Gefährdungen durch Wasserstoff. Auf lange Sicht wird das installierte Wasserstoff-Rekombinatoren-System auch durch Reduzieren des Wasserstoff- und Sauerstoffgehalts zur Inertisierung des Sicherheitsbehälters beitragen. Dieses System ist zwar für Auslegungsstörfälle gedacht, da es jedoch passiv arbeitet, funktioniert es auch bei Schweren Unfällen. Dieser Vorgang ist jedoch langsam und für Schwere Unfälle wäre es vorteilhafter, richtig angeordnete Passive Wasserstoffrekombinatoren (PARs) von höherer Leistungsfähigkeit zu haben.

Die tschechische Strategie beinhaltet:

a) Frühes beabsichtigtes Wasserstoff-Abfackeln (d.h. durch geplante Betätigung von Einrichtungen zu versuchen, eine Deflagration auszulösen), was helfen würde, die Bildung von sensitiven Wolken während der Unfallphase im Druckbehälter zu verringern;
b) Sich-Verlassen auf die Passiven Wasserstoffrekombinatoren (PARs), um den Wasserstoffvorrat im Sicherheitsbehälter allmählich zu verringern;

c) Langzeit-Inertisierung des Sicherheitsbehälters mit Dampf während der Unfallphase außerhalb des Druckbehälters mit Hilfe von Prozeduren zur Steuerung der Sicherheitsbehälter-Sprüheinrichtung, um Zünden durch De-Inertisierung zu vermeiden; und
d) Wenn erforderlich, Gasabgabe aus dem Sicherheitsbehälter über eine Hochdruckentlüftungsleitung und Filter über den Abgaskamin der Anlage, um Wasserstoff aus dem Sicherheitsbehälter abzulassen.

Sowohl tschechische als auch PN7-Berechnungen haben gezeigt, dass es im Fall einer nicht geplanten Betätigung der Sicherheitsbehältersprüheinrichtung, wenn die Wasserstoffkonzentration im Sicherheitsbehälter am höchsten ist, zum Verlust der Integrität des Sicherheitsbehälters kommt. Die tschechischen Unterlagen bieten eine Auswertung der radiologischen Folgen eines solchen Szenarios. Die für Temelín vorgeschlagene SAM Strategie beinhaltet jedoch das Thema einer Gefährdungsminimierung bei einem späteren Versagen des Sicherheitsbehälters auf Grund von Wasserstoffverbrennung, in Übereinstimmung mit den Westinghouse SAMG Vorgaben. Als letzte Maßnahme kann in Temelín das gefilterte Abblasesystem aktiviert werden, das für das Abblasen nach der Sicherheitsbehälterprüfung installiert ist, und so der Sicherheitsbehälterdruck oder der Wasserstoffgehalt reduziert werden. Dieser Problemkreis ist aber anscheinend noch nicht abgeschlossen. Da die Aufheizung der Filter durch den Eintrag von Spaltprodukten zur Erhöhung der Filtertemperatur beitragen kann (einhergehend mit einer Verminderung der Filterwirkung) oder im schlimmsten Fall einen Filterbrand verursachen kann, sollte die Frage der "gefilterte Druckentlastung" in Temelín weiter beobachtet werden.

Die wesentliche Gefährdung bei einem schweren Unfall besteht in der Möglichkeit des Durchbrechens der Bodenplatte.

Die Maßnahmen, die in Temelín für den Fall eines Niederdruckversagen des Sicherheitsbehälters geplant sind, stellen sicher, dass das Einwirken von geschmolzenem Corium auf den Bodenplattenbeton (MCCI) zeitlich hinausgezögert wird. Obwohl diese Maßnahmen in die richtige Richtung gehen, kann nicht mit Sicherheit gesagt werden, ob sie den Schutz der Bodenplatte gegen Durchschmelzen mit geschmolzenem Corium tatsächlich sicherstellen, sollte es zu einem Versagen des Reaktordruckbehälters kommen. Die Eintrittswahrscheinlichkeit für Reaktordruckbehälterversagen ist, wie vor kurzem in einer Analyse gezeigt wurde, zwar gering, existiert allerdings dennoch. Gemäß den Erklärungen der tschechischen Experten sehen die für Temelín geplanten Maßnahmen ein Ausbreiten des Corium und Kühlen mit Wasser vor, was gemeinsam mit einer geplanten, ferngesteuerten Öffnung der Türe im Reaktorschaft den Schmelzfortschritt des Corium aufhalten sollte.

Die Berechnungen, welche im Rahmen des PN 7-Projekts durchgeführt wurden, haben bestätigt, dass die Ausbreitung des Corium das Aufschmelzen verlangsamt und zusätzliche Zeit gewonnen wird. Die Leistungsfähigkeit der Wasserkühlung wurde innerhalb des PN 7 nicht untersucht, da die neuesten Versuchsergebnisse der OECD nicht verfügbar waren. Gemäß neuen Informationen über die Ergebnisse von Großversuchen zum Durchschmelzen von Beton durch geschmolzenes Corium, welche im OECD Programm „The Melt Coolability and Concrete Interaction“ durchgeführt wurden, ergaben, dass in Großversuchen in den Vereinigten Staaten eine verbesserte Kühlung durch Langzeit-Wasser-Kühlung der geschmolzenen Kernschmelzen erzielt wurde. Andere Versuchsergebnisse in diesem Zusammenhang aus Deutschland weisen auf Einschränkungen der Abminderung des Kernschmelze-Angriffs durch Kühlung der Oberfläche der freigesetzten Corium-Schmelze mit Wasser hin. Tschechien nimmt aktiv an einigen Programmen teil und die aktuellen Informationen aus dem OECD-MCCI Programm sind dort verfügbar.

Derzeit kann das Aufhalten des Kernschmelze-Erosionsprozesses nicht eindeutig demonstriert werden. Deswegen überlegen die Temelín-Mitarbeiter zusätzliche Maßnahmen, um die Leckdichte der unter der Bodenplatte liegenden Räumlichkeiten zu gewährleisten. Die Gefährdungspotenziale durch radioaktive Freisetzungen im Fall eines Durchschmelzens der Bodenplatte sind viel geringer als dies bei einem frühzeitigen Versagen des Sicherheitsbehälters der Fall wäre. Wie an anderer Stelle gezeigt wurde, gilt für Freisetzungen bei Bodenplattenversagen, dass die Menge an radioaktiven Aerosolen, die sich weiterhin in der Atmosphäre befinden, im Vergleich mit frühzeitigem Sicherheitsbehälterversagen dramatisch (um Größenordnungen) vermindert ist . Wenn man das Wiederverflüchtigen und das Abdampfen von schon abgelagerten Kontaminantien bei spätem Sicherheitsbehälterversagen außer Acht lässt, wird die radiologische Gefährdung der Umgebung in gleicher Weise reduziert.

Während des Expertentreffens in Prag haben die tschechischen Experten in Beantwortung von Fragen die Umweltparameter im Reaktorgebäude nach dem Durchschmelzen der Bodenplatte diskutiert. Die tschechischen Experten sprachen über eine in Entwicklung befindliche Strategie, die das Wiederverflüchtigen von bereits abgelagerten Spaltprodukten verhindern helfen soll. Ursache dafür könnten starke Gasturbulenzen im Sicherheitsbehälter bei der Druckentlastung durch Durchschmelzen der Bodenplatte sein. Die in Entwicklung befindliche Strategie schließt auch die Verhinderung von Wasserstoffverbrennung im Reaktorgebäude nach dem Durchschmelzen der Bodenplatte ein. Damit soll die Integrität des Reaktorgebäudes erhalten werden, um durch den natürlichen Aerosolabsetzmechanismus den Quellterm zu reduzieren. Zusätzlich können dann Teile des Gasgehalts aus dem Reaktorgebäude über den Abgaskamin (via Reaktorgebäudelüftung) freigesetzt und so eine größere Verdünnung und geringere Strahlendosen für die Umwelt erzielt werden.

Diese Strategie würde eine Druckentlastung des Sicherheitsbehälters vor dem Durchschmelzen der Bodenplatte über das Entlüftungssystem (Hochdruckleitungsanschluss am Abgaskamin) beinhalten. Dadurch würde auch die Wasserstoffkonzentration im Sicherheitsbehälter verringert. Die tschechischen Experten haben diese Fragen im Rahmen des Expertentreffens in Prag erwähnt, jedoch wurden keine detaillierten Informationen über den verfolgten Ansatz zur Verfügung gestellt.

Die Maßnahmen und Strategien zur Verminderung von Spaltproduktfreisetzungen stehen im Einklang mit der internationalen Praxis. Die offenen Fragen sind vor alllem mit der Verminderung der radioaktiven Freisetzungen verbunden, die im Fall des Durchschmelzens der Bodenplatte durch die Kernschmelze auftreten. Die tschechischen Experten sehen darin ein Problem, das später behandelt wird, während sie als vordringlichste Aufgaben diejenigen sehen, die die Verhinderung des Durchschmelzens der Bodenplatte zum Inhalt haben.

V. Empfehlungen für ein weiterführendes Monitoring

Der Monitoring-Vorgang, wie er bisher im Rahmen des „Brüsseler Abkommens“ (ANNEX I) abgewickelt wurde, hat im Bereich der schweren Unfälle dazu beigetragen, einige der relevanten Fragen zu klären. Es wurde gezeigt, dass in Temelín ein umfangreicher Prozess stattfindet mit dem Ziel, ein umfassendes Management für Schwere Unfälle und die Vermeidung von Folgewirkungen aus schweren Unfällen zu implementieren. Dieser Prozess ist jedoch noch im Laufen und das Experten-Team kann sich zur Zeit nur an einer Anzahl von Ansichten und Erwartungen zur endgültigen Einführung des Managements für Schwere Unfälle orientieren, wie diese von tschechischer Seite vorgebracht wurden.

Das Experten Team empfiehlt der Österreichischen Bundesregierung, folgende Erkenntnisse im Rahmen des laufenden bilateralen Übereinkommens zwischen Österreich und der Tschechischen Republik weiterzuverfolgen.

Folgenden Bereiche sind von Interesse:

- Die unterstützende Untersuchung zu schweren Unfällen und die Probabilistische Sicherheitsanalyse, sowie deren Verwendung zur Verifikation von Managementstrategien bei schweren Unfällen und damit zusammenhängende Vorgangsweisen
- Die Arbeiten zur Einführung von Richtlinien zum Management von schweren Unfällen (SAMG) unter Einschluss der Vorgangsweise, des Anwendbarkeitsnachweises (für SAMG) und der Unterweisungen der Betriebsmannschaften in Bezug auf das Management von schweren Unfällen
- Bestimmung des zulässigen Ausmaßes von Inhomogenität der Wasserstoffverteilung in der Atmosphäre
- Einführung von Anlagenänderungen zur Verbesserung von technischen Maßnahmen für das Management von schweren Unfällen

Eine etwas detailliertere Erörterung der vorgeschlagenen Monitoring Fragestellungen aus diesen Sachgebieten wird in der Folge gegeben.

Das Experten-Team empfiehlt der Österreichischen Bundesregierung, die Weiterführung des Monitoring von Berechnungen, die in Temelín unter Verwendung des Programms MELCOR 1.8.5 und anderer Rechenprogramme in Zukunft durchgeführt werden, in Betracht zu ziehen, um detailliertere und verifizierte Informationen zu folgenden Fragen zu erhalten:

- Leistungsfähigkeit der PORV, gemeinsam mit der Leistungsfähigkeit der geplanten Druckentlastung des Kühlmittelsystems;
- Genehmigungsvorschriften für die Wasserstoffüberwachung und deren Leistungsfähigkeit und/oder zusätzliche Anwendung von gefiltertem Gasablassen zur Minderung der radioaktiven Freisetzung ;
- Betriebliche Kapazitätscharakteristik des Notabgassystems unter den Bedingungen von schweren Unfällen;
- Untersuchungen zum Versagen der Bodenplatte durch Durchschmelzen.

Des weiteren empfiehlt das Experten-Team der Österreichischen Bundesregierung, die Maßnahmen zur Einführung der Richtlinien zum Management von schweren Unfällen in Temelín weiterhin zu beobachten, um sicherzugehen, dass die verbleibenden Schritte des Implementierungsprozesses erfolgreich abgeschlossen werden. Weitere wichtige Problemkreise, die zu verfolgen bzw. zu verifizieren wären, umfassen die überarbeiteten Prozeduren, SAMG Validierung und die Ausbildung des Personals. Gleichzeitig sollte von Seiten der österreichischen Regierung den Empfehlungen aus jeder unabhängig durchgeführten Überprüfung des Managements von schweren Unfällen und deren Umsetzung volle Aufmerksamkeit entgegengebracht werden.

Technische Maßnahmen, die für die Vermeidung und Einschränkung von risikoträchtigen Szenarien erforderlich sind, sollten überprüft werden, um den Nachweis zu erbringen, dass in der Anlage die richtigen Vorkehrungen getroffen wurden (das gilt sowohl für Prozeduren als auch für Anlagenänderungen). Angemessene Aufmerksamkeit sollte schweren Unfällen entgegengebracht werden, die ein besonderes Sicherheitsrisiko darstellen, wie etwa das Durchdringen der Bodenplatte bei Austreten von Kernschmelze aus dem Reaktordruckbehälter und ein Totalausfall der Stromversorgung. In diesem Zusammenhang erwähnenswerte Aspekte beinhalten Maßnahmen zum zeitgerechten Öffnen der Türe im Reaktorschaft vor dem Versagen des Reaktordruckbehälters, ebenso wie den Schutz der Sicherheitsbehälterdurchdringungen und der Dichthaut des Sicherheitsbehälters gegen Kernschmelze-Beton-Reaktionen (MCCI), sowie auch die Erweiterung der Batteriekapazitäten. Weiterführende analytische Arbeiten, die von der Anlage und den Technischen Support Firmen zu den Kernschmelze-Beton Reaktionen durchgeführt werden und zur Begrenzung der damit in Verbindung stehenden radiologischen Folgewirkungen führen sollten, wären ebenfalls in das weiterführende Monitoring aufzunehmen.

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1 INTRODUCTION

1.1 Background of the project

The Republic of Austria and the Czech Republic have, using the good offices of Commissioner Verheugen, reached an accord on the “Conclusions of the Melk Process and Follow-up” on 29 November 2001. In order to enable an effective use of the “Melk Process” achievements in the area of nuclear safety, the Annex I of this “Brussels Agreement” contains details on specific actions to be taken as a follow-up to the “trialogue“ of the “Melk Process” in the framework of the pertinent Czech-Austrian Bilateral Agreement.

To enable an effective ”trialogue” follow-up in the framework of the pertinent Czech-Austrian Bilateral Agreement, a seven-item structure given in Annex I of the “Brussels Agreement” has been adopted. Individual items are linked to:

- Specific objectives set in licensing case for NPP Temelin units;
- Description of present status and future actions foreseen by the licensee and SÚJB respectively.

Each item under discussion will be pursued according to the work plan agreed at the Annual Meeting organised under the pertinent Czech-Austrian Bilateral Agreement.

Furthermore, the Commission on the Assessment of Environmental Impact of the Temelín NPP - set up based on a resolution of the Government of the Czech Republic - presented a report and recommended in its Position the implementation of twenty-one concrete measures (Annex II of the “Brussels Agreement).

The signatories agreed that the implementation of the said measures would also be regularly monitored jointly by Czech and Austrian experts within the Czech-Austrian Bilateral Agreement.

A “Roadmap” regarding the monitoring on the technical level in the framework of the pertinent Czech-Austrian Bilateral Agreement as foreseen in the “Brussels Agreement” has been elaborated and agreed by the Deputy Prime Minister and Minister of Foreign Affairs of the Czech Republic and the Minister of Agriculture and Forestry, Environment and Water Management of the Republic of Austria on 10 December 2001.

This „Roadmap“ is based on the following principles:

- *The implementation of activities enumerated in Annex I and II of the “Brussels Agreement” will be continued to ensure that comprehensive material is available for the monitoring activities set out below.*
- *Having in mind the peer review procedure foreseen by the EU to monitor the implementation of the recommendations of the AQG/WPNS Report on Nuclear Safety in the Context of Enlargement, the Czech and Austrian sides agree that this peer review should serve as another important tool to handle remaining nuclear safety issues.*
- *As a general rule the regular annual meetings according to Art. 7(1) of the bilateral Agreement between the Government of Austria and the Government of the Czech Republic on Issues of Common Interest in the Field of Nuclear Safety and Radiation Protection will serve to monitor the implementation of those measures referred to in Chapter V of the Conclusions and to address questions regarding nuclear safety in general, in particular those issues which – according to Chapter IV of the Conclusions - have been found, due to the nature of the respective topics, suitable to be followed-up in the framework of this - Bilateral Agreement.*
- *In addition, specialists’ workshops and topical meetings will take place, organised as additional meetings according to Art. 7(4) of the bilateral Agreement between the Government of Austria and the Government of the Czech Republic on Issues of Common Interest in the Field of Nuclear Safety and Radiation Protection, as set out in the “Roadmap”.*

The Federal Ministry of Agriculture, Forestry, Environment and Water Management entrusted the Umweltbundesamt (Federal Environment Agency Ltd. with the general management of the implementation of the “Roadmap”. Each entry to the “Roadmap” corresponds to a specific technical project. [see ANNEX C].

The objective of the Roadmap process covered by the item 7 as stated in Annex I of the “Brussels Agreement” is: *"Effective prevention and mitigation of consequences of beyond design basis accidents (severe accidents)”.*

Annex I provides the following statements regarding the *“Present Status and specific Actions Planned”:*

“A set of preventive and mitigative measures is, at present, applied in NPP Temelín with respect to beyond design basis accidents. These include software and hardware measures, among others, e.g. Symptom Based Emergency Operating Procedures, Technical Support Centre, Post Accident Monitoring System, Emergency Preparedness.

For the purpose of emergency preparedness, the PSA was employed with the aim to identify and group events with different initiating occurrences, but with similar end-effects. On the basis of this assessment the relative risk was estimated for specific events in order to select those, which will serve for the determination of emergency response activities (pre-planned, reactive).

Severe Accidents Management Guidelines (SAMG) as a state-of the-art tool will complete the whole system of mitigation measures with respect to the beyond design basis accident management. The project for SAMG development is scheduled to be finished by end 2002 to be followed by validation.

To foster mutual understanding two lines of activities will be followed within the framework of the bilateral agreement:

a) *A Working Group on comparison of calculations regarding the radiological consequences of BDBA with a view to harmonise the basis for emergency preparedness will be established.*
b) *The exchange of information related to SAMG will include discussion on the analytical basis as well as on corresponding software and hardware measures."*

The issue (a) has been covered by a separate project PN1 [see ANNEX C], the issue (b) is covered by this project .

Refering to Chapter IV of the “Brussels Agreement” and the principles of the “Roadmap”, a number of issues identified in the “trialogue” of the Melk Process are found suitable to be followed-up in the framework of the Bilateral Agreement. The following seven issues are closely related to the topic of item No. 7 (b) and are therefore also covered in this project:

- Issue No. 1 Containment bypass and preliminary-to-secondary (PRISE) leakage accidents
- Issue No. 4 Containment Design and Arrangement
- Issue No. 5 Probabilistic Safety Assessment and Severe Accidents
- Issue No. 6 Emergency Operating Procedures EOPs & Severe Accident Management Guidelines (SAMGs)
- Issue No. 16 Hydrogen Control
- Issue No. 26 Beyond Design Bases Accident Analysis
- Issue No. 29 Technical Basis for Temelín Emergency Planning Zones (EPZs)

The Roadmap specified that the related Specialists' Workshop would be held in the 1st half of 2003 to discuss this issue. This workshop on the "Roadmap" item No. 7 was conducted in Prague on 17 and 18 June 2003 according to Article 7 (4) of the Bilateral Agreement of the Exchange of Information on Nuclear Safety. This workshop was the key element in the monitoring process. The analysis of information made available there played a significant role in the development of the basis for the Preliminary Monitoring Report.

A Specialists' Team of international experts was committed by the Umweltbundesamt (Federal Environment Agency) on behalf of the Austrian Government to provide technical support for the monitoring on the technical level of the implementation of the SAMGs Issue as listed in Annex I of the Conclusions of the Melk Process and Follow-up. This specific technical project is referred to as project PN7 comprising altogether seven predefined "project milestones" (PMs).

In a series of presentations the outline of the technical approach to the Severe Accidents Management Guidance (SAMG) item was described by Czech experts, including the legal framework for the issue and the information provided to the Licensing Authority about the technical approach.

The Czech presentations at the Specialists' Workshop covered a broad scope of aspects related to the development and implementation of Symptom Oriented Emergency Operating Procedures (EOPs) and SAMGs. The following presentations were provided:

- Regulatory Approach to Accident Management
- National Projects and International Programmes in Accident Management
- The Temelín Accident Management Programme
- Temelín Probabilistic Safety Assessment (PSA) Update Results
- VVER-1000 PSA Studies Comparison
- Progress in Analytical Tools for Severe Accident Analysis
- Existing Severe Accident Analyses for Temelín
- Temelín EOPs - Severe Accident Prevention
- Westinghouse Owners Group (WOG) Approach to SAMGs
- Temelín SAMGs Development & Implementation
- Concluding Positions of Temelín NPP and SÚJB

The approach to Severe Accident Management of the Temelín Nuclear Power Plant is to rely on a systematic process, which has been established and used by the Westinghouse Owners Group (WOG) for the development, implementation and maintenance of Severe Accident Guidelines.

The presentations provided insight into the extensive work accomplished by the plant operator and its technical support organisations to consolidate the SAMGs in the framework of the implementation of emergency procedures.

The descriptions identified the approach taken, but as overviews they provided only limited insight into the results and how these were obtained. A number of the questions posed by the Specialists' Team were considered by the Czech side to exceed the level of detail or the scope of the Roadmap Workshop activities. Consequently, both sides agreed that the pertinent Czech-Austrian Bilateral Agreement is the appropriate framework giving the opportunity for further discussion and sharing additional information on these issues.

From technical point of view the assessment of SAM addresses all the elements and aspects, which are recognized important in the preparation and development of SAM program. These include supporting accident analysis, assessment of plant vulnerabilities, selection of AM strategies, evaluation of plant equipment and instrumentation, development of accident management procedures and guidelines (SAMGs), and their implementation (including the SAMG verification and validation, related staffing and qualification, training, and co-ordination with emergency preparedness plan).

The project includes all related activities such as the identification of information sources for plant specific data, which are needed for the assessment, analysis of the reference material provided by the plant, and evaluation of the current plant status against the state-of-the-art practice.

Gathering appropriate information on the plant status with regard to the above-indicated areas is an essential part of the project. The main concept implemented in the project was to break down the overall subject into the line items, which could then be verified for completeness and compliance with the accepted international practice. They are further called Verifiable Line Items (VLIs).

The first step of the project (project milestones 1) focused on the definition of VLIs. This task was the “road map” for the whole project. The VLIs were identified considering both the state-of-the-art practice in the subject and the available knowledge on the plant status. Information on the plant status was gathered from the technical documents and publications on the Temelín NPP, previous studies conducted within the framework of ‘Melk Process’, and the results of accident calculations conducted by the project team for the Temelín NPP as well as the available results for similar plants.

The second step (project milestones 2), the Specifique Information request (SIR) considered to contain the kind of information required to provide profound answers to the VLIs.

The third step (project milestones 3) was intended to complete all the preparatory activities for the workshop (PM 3). This included the benchmarking of information/documents provided by the Czech side during the workshop against the state-of-the-art consolidated practice. VLIs formulated in the Task 1 were used for this purpose. The scope of this task also included the development of briefing material and preparation of the briefing session for the Austrian delegation.

Project milestones includes the preparation of a preliminary monitoring report (PMR) on the status of SAM at Temelín (PM 4). This task was conducted based on the results of the Prague workshop. This report is also intended to identify SAM related issues for further monitoring by Austrian Government.

At the time of the Specialists Workshop, the SAMGs were in the process of translation by the operator for training and implementation in 2004. Further monitoring should therefore focus in some detail on the SAMG implementation process, including further attention to some specific plant design changes (changes to primary coolant system depressurization options, core debris cooling procedures, core debris spill management, etc.), announced at the Specialists Workshop, as part of SAMG implementation.

Further tasks (project milestones 5 – 7) will concentrate on the consolidation of findings and their presentation in the form of final reports (Final Monitoring Report and Summary Monitoring Report).

1.2 Technical Background

The Temelín nuclear power plant (NPP) is a two-unit facility originally designed as WWER-1000/320 pressurised water reactors according to the standards of the former Soviet Union. Following the "Velvet Revolution", the plant design was upgraded (including implementation of fuel and instrumentation & control equipment delivered by Westinghouse) and placed into operation beginning with Unit 1, which began startup testing in 2001.

NPPs are designed based on the occurrence of certain initiating events such as loss of coolant accidents (LOCAs), transient events (such as loss of offsite power or loss of feedwater), and external man-made and natural phenomena hazards (such as fires, earthquakes, flooding, and the like). In plant safety analyses reports (SARs), the plant response to such "design basis accidents" (DBAs) is evaluated assuming a single active failure in the safety system response, and the performance of the plant is evaluated to ensure that basic safety criteria are met. Such SAR assessments are performed with a substantial measure of conservatism, including the assumption of a degree of core damage that is far out of proportion to the actual circumstances, which are most likely to exist in reality. Prior to the end of the 1970s, it was widely believed that the likelihood of a severe accident involving actual core damage in commercial NPPs was extremely low (i.e., less than one in a milion per year - that is, a frequency less than 10^{-6} per year).

Although severe accidents had been studied for some time already (including the first quantitative risk assessment of severe accidents in the 1975 "Reactor Safety Study", WASH-1400), the issue of severe accidents gained increased international significance following the severe accidents at the Three Mile Island Unit 2 reactor in 1979 and the Chornobyl Unit 4 reactor in 1986. As a consequence of these accidents, increased application of probabilistic safety assessment (PSA), and the results of international research and development programmes, the issue of accident management (AM) gained prominence in the 1990s.

Severe accident management (SAM) has a formal, internationally recognized definition and framework [NEA 96]: "*SAM consists of those actions that are taken by the plant staff during the course of an accident to prevent core damage, terminate progress of core damage and retain the core within the vessel, maintain containment integrity, and minimize off-site releases. SAM also involves pre-planning and preparatory measures for SAM guidance and procedures, equipment modifications to facilities procedure implementation, and severe accident [management] training. The overall objective is to further reduce the risks of large releases. It is the responsibility of the licensees to develop and implement a SAM program.*"

SAM plays an important role in the defence-in-depth concept for accidents, which exceed the design basis. SAM and so-called "complementary measures" define Level 4 of defence-in-depth, which represents an attempt to bring severe plant conditons under control and/or to mitigate a radioactive effluents release which may nonetheless occur despite the capabilities of plant systems and structures and the best efforts of the plant staff. "Complementary measures" are defined as plant systems, structures, or components (SSCs) which have been added to the design of the plant beyond the normal complement of SSCs required to prevent or mitigate design basis accidents (DBAs), and whose function is related to severe accident prevention or to the mitigation of the consequences of severe accidents (that is, related to accident management).

The last level of defence-in-depth, Level 5, consists of the implementation of pre-planned off-site emergency response measures to reduce the consequences of severe accidents.

In the context of the Temelín Road Map, the issue of severe accidents and severe accident management was raised during the Melk Protocol process. When the Czech Republic and the Republic of Austria jointly issued the Melk Concluding Statement and the Road Map, the issue of severe accidents and their management was specified for further technical exchange.

1.3 Technical Approach – The Horizontal and Vertical Approaches

Two teams under the Technical Project Management of Enconet and IRR-ARCS performed the monitoring work on this project.

The Enconet team addressed the broad general assessment of Temelín SAM and SAMGs based on underlying analyses and principles. This approach provided a broad ‘horizontal’ view of SAMGs and SAM.

The IRR-ARCS team addressed one of the possible impairments of the WWER 1000 containment function of the Temelín NPP due to adverse hydrogen combustion behaviour. Investigations in that area, which focuses on in-depth analysis of the specific vulnerability of the Temelín containment, are further called the 'vertical evaluation'.

The technical work was managed separately and in parallel without interference but with a project Synergy Group providing transfer of information and joint discussion of important issues. The PMR is the joint responsibility of Enconet and IRR-ARCS.

1.4 Assessment Framework

The assessment of the effectiveness of the prevention and mitigation of severe accidents at Temelín has been performed in the context of several activities. In the main, these activities have involved an assessment of the state-of-the-art in severe accident management in Western Europe (and more broadly including the US), an assessment of the performance of a generic WWER 1000 NPP in severe accidents (with consideration of Temelín-specific plant characteristics to the extent possible derived from available information to the extent possible) by means of state-of-the-art code calculations, and a two-day Specialist Workshop held in Prague at which presentations on various aspects of accident management were made by Czech experts and discussed with these experts by an Austrian delegation which included members of the PN7 Technical Project Management Teams.

It should be clear from the outset that the assessment is not based on a review of original Temelín documentation. That is, the on-site and off-site emergency response plans, the Emergency Operating Procedures (EOPs), and the Severe Accident Management Guidelines (SAMGs) themselves were not provided for review. Similarly, although a number of severe accident progression calculations were performed by Czech experts as technical support for the development of SAMGs, the details of these calculations were also not provided for review. Likewise, the updated Probabilistic Safety Assessment (PSA) and Pre-Operational Safety Analysis Report (POSAR) were not available for review. Finally, the plant itself was not available for detailed confirmation of geometric arrangement and other details.

It should also be clear, however, that the PN7 Technical Project Management Teams from ENCONET and IRR-ARCS did not come to project with a blank sheet of paper. Both Teams made use of experienced project personnal and subcontractors with familiarity with EOPs and SAMGs based on the Westinghouse approach, and with experience in severe accident progression calculations. Both Teams have previously reviewed the POSAR and the PSA documentation, and have had the opportunity to discuss severe accident issues with Czech experts over the past three years in which the Melk Protocol and Road Map activities have taken place.

Thus, the PN7 approach is to capitalize on this experience, on accident progression calculations made with state-of-the-art codes, on knowledge of the state-of-the-art in accident management in Western Europe and the United States, and on knowledge gained over the years about the Temelín plant design and its risk profile (as well as broad and in-depth knowledge of severe accident and accident management issues across the industry in Europe and elsewhere), to make an informed judgment about the prevention and mitigation of hypothetical severe accidents which might occur at Temelín.

One basic fact is that rules and regulations applied to Temelín NPP were combined for the design, construction and design verification as well as for the introduction of SAMGs from elements from rules and regulations originating from Russian, Czech and United States licensing authorities.

1.5 Structure of the report

Sections 2 – 5 provide a comprehensive evaluation of relevant aspects relating to severe accident management (SAM) and SAM Programme at Temelín. The material presented in these sections is arranged into several subsections corresponding to the selected evaluation factors or aspects.

With some exceptions, each of these subsections comprises of three parts: *'The current state-of-the-art requirements and practices', 'Current plant status'* and *'Evaluation'*. The first part provides 'assessment criteria' specific to the evaluation area/factor i.e. the basis to be used for the assessment. Typically, the *'Current plant status'* part includes a brief discussion of the related plant status with references to relevant documents or other sources of information. The *'Evaluation'* part summarizes the results of the assessment against the 'specific assessment criteria'. Deficiencies or safety concerns as well as issues for further monitoring are identified.

Section 2 provides overview of the general approach to SAM and SAM Programme at Temelín and interfaces with the emergency response plan.

Section 3 presents a more detailed evaluation of SAM at Temelín. Aspects addressed in this section include the supporting analyses (deterministic and probabilistic), development of SAM strategies, performance of equipment in severe accident conditions, SAM related arrangements for personnel response, contents and structure of SAM Guidelines (SAMGs) and the process for SAMG development and implementation.

Section 4 provides comprehensive discussion of SA phenomenology. The accident analyses conducted within the PN7 project are described with the rationale for their selection and a brief overview of the results. This section provides relevant background information for the evaluation of Temelín plant specific SAM strategies.

Evaluation of SAM strategies adopted for the Temelín plant is provided in Section 5. Various threats to the containment integrity and proposed SAM measures are discussed.

Section 6 presents the main conclusions and recommendations for further monitoring by the Austrian Government.

Annex A summarizes the results of comparison study on general approaches to SAM in Western Europe and the USA.

Annex B provides detailed information on SA calculations that were used to support the evaluation of SAM at Temelín. These include the simulations conducted within the PN project as well as the existing (published) analyses conducted elsewhere.

2 SAM APPROACH VERSUS REQUIREMENTS AND EMERGENCY RESPONSE INFRASTRUCTURE

2.1 National Requirements

VLI No.	VLI title / description
1.1.1	Are there any national requirements on the overall plant-specific AMP (Accident Management Program) and SAMP (Severe Accident Management Program)?
1.1.2	Have national requirements been addressed in the plant AMP?
1.1.3	Have national requirements been addressed in the plant SAMP?

State-of-the-art requirements and practices

According to the current state-of-the-art safety philosophy [IAEA 2000a], the consideration of severe accidents in NPPs includes the identification of event sequences that lead to severe accidents, consideration of existing plant capabilities to return the plant to a controlled state and to mitigate the consequences of the severe accident, evaluation of potential design changes which could either reduce the likelihood of these events or would mitigate the consequences, and establishing accident management procedures.

A set of actions taken during the evolution of an event sequence towards beyond design basis accident (BDBA), which is known as Accident Management (AM), is intended to prevent the escalation of the event into a severe accident, to mitigate the consequences of a severe accident, and to return the plant to a controllable safe state. The preventive actions are covered by the Emergency Operating Procedures (EOPs). The mitigative actions are addressed within the Severe Accident Management Guidelines (SAMGs).

Accident Management Programmes (AMP) based on this concept were adopted in many nuclear plants starting from early 1980s. AMP comprises plans and actions undertaken to ensure that personnel with responsibilities for AM are adequately prepared to take effective on-site actions to prevent or to mitigate the consequences of a severe accident and, when deemed necessary, to plan and implement plant modifications.

Typically, the Severe Accident Management is considered as part of and incorporated into the on-site Emergency Response Plan (ERP). The requirements concerning the emergency preparedness of the plant are normally part of the national law. However, while many Western-European regulatory authorities required the development and implementation of SAMGs (industry commitments to develop and implement SAMGs are also widespread), in most cases the SAMGs are not formally approved by the regulatory authority. Many Western European utilities addressed SAM issue as a voluntary industry initiative and follow the general recommendations of international organizations.

Regulatory requirements were indeed put forward in Belgium, the Netherlands, Sweden, Finland, Germany (as a binding RSK recommendation), Switzerland, the United Kingdom (implicit in the regulations on tolerable risk), and in France. Of the Western European nations with nuclear power plants, the only exception to this is Spain, which strictly follows the USNRC position which accepted an industry commitment in lieu of development of regulatory requirements or issuance of some regulatory vehicle such as a Generic Letter.

A more detailed discussion of SAM related practices in the EU countries and the USA can be found in Annex A.

The obligations, responsibilities, criteria and requirements for emergency response planning are set up in the IAEA series of international safety standards [IAEA 82, IAEA 2000 a - c, IAEA 94, IAEA 96]. Similar requirements are provided in US NRC guides [NUREG-0654, NUREG -0696, NUREG -0818].

Current plant status

The Czech Regulator (SUJB) has initiated in 1994 a systematic programme on Accident Management [Miasnikov 03]. Internationally recognized safety guidelines published in documents prepared by the IAEA [INSAG-10, INSAG-12, IAEA 88] and the US NRC [NRC 88, NRC 90] provided a basic input for the development of this programme. The programme was adopted at the plant and is being implemented on voluntary basis. SUJB provides also a technical support for the TSO in SA related research.

The SUJB approach focuses on the definition of acceptance criteria and providing guidance leaving flexibility on the selection of potential AM actions. This approach is consistent with the existing legal framework in Czech Republic requirements on emergency preparedness [Miasnikov 03].including the

Symptom based Emergency Operating Procedures (EOPs) have already been implemented at Temelín based on the generic Westinghouse Emergency Response Guidelines [WEC 83]. These procedures have been prepared in accordance with the current international practice. They have been subject to extensive verification and validation using the plant specific simulator [Hončarenko 03].

Work on the implementation of Severe Accident Management Guidelines at Temelín is well advanced. Westinghouse has developed the SAMG package in accordance to the WOG SAMG generic guidelines [WOG 94, WOG 01]. Currently this package is being translated into Czech language and subjected to internal review. Work on the preparation of SAMGs validation using the plant specific simulator is underway [Dessars 03, Sỷkora 03 b]. It should be noted that a Technical Support Centre, which is intended to implement the new SAM functions in case of a severe accident, was established earlier as part of the implementation of symptom based EOPs. Its organization structure and staffing will be altered to accommodate the implementation of SAMGs.

Evaluation

The approach to SAM being implemented in Czech Republic for the Temelín plant reflects the current state-of-the-art in this area. Legal framework for the Czech National Emergency Preparedness Program was clearly presented by SUJB and Temelín experts during the Prague workshop [Miasnikov 03]. Requirements on the content of off-site emergency plan and emergency rules were discussed also. It was stated that the existing Temelín-specific on-site E-plan covers the classification of emergency situation severity, personnel duties and responsibilities as well as the emergency facilities and responses. These are the most important areas that should be covered in such a document. Specific regulations issued by SUJB on this subject have been mentioned in this context [Miasnikov 03].

As stated in the presentation [Miasnikov 03], the E-plan, which includes SAM-related interfaces, is being reviewed by SUJB. It can be expected that the existing national requirements on the emergency response planning will be fulfilled and the plan will be properly adjusted to accommodate SAM. It should be noted that details of the related emergency response administrative framework at Temelín plant were not presented at the Prague workshop and it is not possible to evaluate its contents and quality. However with regard to the SAM related interfaces the current good practices are followed. Some concerns regarding the required authorization of possible plant intentional release from the containment can be raised (see Section 2.3).

2.2 On-site ER Infrastructure and its Consistency with the Overall ER Plan

VLI No.	VLI title / description
1.2.1	Has the overall organisation for emergency response been clearly defined?
1.2.2	Has the overall emergency preparedness been subject to independent / external review
1.2.3	Are all the parts of the on-site emergency response infrastructure (MCR, TSC, Operation Support, etc.) addressed in the overall emergency plan?
1.2.4	Have the existing emergency response procedures been reviewed to address changes relating to SAM

State-of-the-art requirements and practices

The organisation of emergency response should be clearly described in the ERP of the operating organisation. The related state-of-the-art requirements are defined in the IAEA safety standards [IAEA 2000 a, IAEA 02]. A brief overview of requirements regarding ERP is provided below.

The ERP should describe the on-site organisation used to perform the emergency response (ER) functions and conditions under which an emergency should be declared, including the criteria for classifying the event, a list of job titles and/or functions of persons empowered to declare it, and a description of suitable arrangements for alerting response personnel and public authorities. Arrangements for minimising the exposure of persons on and off the site to ionizing radiation, and for ensuring medical treatment of casualties should be described.

The plan should define the chain of command and communication as well as the related facilities and procedures. Inventory of the emergency equipment to be kept in readiness at specified locations and actions to be taken by persons and organisations involved in the implementation of the plan for each class of emergency should be described. Arrangements for declaring the termination of an emergency should also be included.

The operator and ER organisations should identify the knowledge, skills, and abilities necessary to be able to perform the emergency functions. The operator and the response organisations should make arrangements for selection of personnel and training to ensure that the personnel have the requisite knowledge, skills, abilities, equipment, and procedures to perform their assigned emergency response functions.

Current plant status

The onsite emergency plan has been revised and submitted for review by SUJB to account for changes arising from implementation of the SAMGs. There was only minor discussion of the E-Plan/SAMG interface during the Prague workshop. [Dessars 03, Miasnikov 03, Sỷkora 03 b]. The activation of Technical Support Centre upon entry into the EOPs and SAMGs was discussesd, along with definition of responsibilities and authorities for using SAMGs, and the transfer of information to off-site authorities. These aspects are considered as an integral part of the SAMG implementation process. It is understood that the existing ER implementing procedures will be reviewed in due time to address changes relating to the implementation of plant specific SAMGs.

Evaluation

It can be concluded that relevant AM related interfaces with the plant specific E-plan are identified and taken into consideration in the development of SAMGs. Plant arrangements and activities in this area are consistent with the current stage of the SAMG implementation process. There is no reason not to expect that the on-site emergency response infrastructure (MCR, TSC, Operation Support, etc.) will be properly addressed with regard to their role, and that the related interfaces will be specified in the plant procedures (administrative procedures, EIPs, EOPs, SAMGs, etc.) and the overall Emergency Response Plan (on-site and off-site). Confirmation of this could be a subject for further monitoring, or could be handled within the area of emergency response organisation (interface between Austria and the Czech Republic).

2.3 Organisation and Responsibilities in SAM

VLI No.	VLI title / description
1.3.1	Is there a clear division of preventive actions (EOPs) and mitigation actions (SAMGs) and associated responsibilities?
1.3.2	Have the personnel needed/involved in AM been specified?
1.3.3	Are the lines of responsibility and authority for all the personnel involved in AM clearly defined?
1.3.4	What are the criteria for transition of responsibilities for AM from the MCR to on-site emergency response centre (TSC)?
1.3.5	Are the available human resources at the plant sufficient for SAM?
1.3.6	Has an adequate call-on system for personnel involved in SAM been established?

State-of-the-art requirements and practices

Typically, the implementation of AM during an accident is based on a clear division of preventive AM actions (covered by EOPs) and mitigation AM actions (covered by SAMGs). As a general rule the main control room (MCR) staff and the Shift Supervisor are responsible for implementation of mitigation actions (within the EOPs) and the TSC staff is responsible for the use of SAMGs. In the domain of EOPs, the MCR would request support from the TSC where it is considered useful. In SAMG domain, all decisions are made in the TSC, where the only formal role of the MCR is the execution of strategies decided upon in the TSC.

The criteria for transition of AM from EOPs to SAMGs should be clearly defined in the plant procedures (EOPs). Typically, there are 3 specific EOPs from which such a transition can take place. In WOG plants these are ‘Response to Inadequate Core Cooling’ (FR-C.1), ‘Loss of All AC Power’ (ECA-0.0), and ‘Response to Nuclear Power Generation/ATWS’ (FR-S.1) [WOG 01].

The lines of responsibility and authority for all personnel involved in AM (MCR and TSC staff as well as of other staff of the emergency response organisations), need to be specified in the plant procedures. One of the important SAMG implementation issues is the authorisation of intentional discharge of radioactivity from the containment and associated responsibilities. This authorisation should be clearly defined in the overall Emergency Response Plan (ERP).

Typically, the main responsibility for overall SAMG evaluation and implementation is assigned to TSC Director (On-site Emergency Director). The TSC Director is supported by the SAMG decision-making support group (DMSG), which co-ordinates and monitors the implementation of the selected SAMG strategies and assesses their effectiveness.

The TSC DMSG group should include a/o the TSC technical support coordinator, radiation protection coordinator, TSC operations coordinator, engineering coordinator, maintenance coordinator, and TSC information coordinator. The TSC technical support coordinator provides supervision to the SAMG evaluation group (EG), which is responsible for the evaluation of plant conditions and SA challenges and for the determination of SAM strategies using SAMGs. TSC support should also include staff responsible for TSC communication, data transfer, computer operation, and core physics support coordination. The TSC is required to be functional within certain time period following an emergency event (in accordance to emergency criteria stated in the ERP). Typically, this should be not be delayed more than 1 hour.

Current plant status

The dividing line between EOPs (preventive actions) and SAMGs (mitigative actions) has been clarified. The division of responsbilities and decision-making authority between the TSC and the MCR has also been clarified. With regard to the authority and responsibilities, focus was put in the workshop presentations on lower level of decision-making (MCR and TSC). Not much has been said relating to the authority/responsibility assigned to higher level staff/positions. It seems that the plant's intentions for taking a decision on an intentional release of radioactivity to the environment (as during filtered venting) is are not consistently clear within the organisation. One senior official stated flatly that this would not be done, however the SAMGs clearly envision it being done. In response to a direct question relating to this issue the Temelín experts answered that on-going strategy will be discussed with local authorities but that there is no written requirements / obligations [Sykora 03 a].

Transition of responsibilities for AM from the MCR to on-site emergency response centre (TSC) is well defined [Dessars 03]. Basic criteria for the transition from the EOPs to SAMGs are defined (‘core exit temperature > 650°C and all recovery actions have failed’). These criteria are consistent with the WOG SAMG generic concept [WOG 01]. At the moment, the current revision of EOPs at the plant does not comply with these transition criteria because Temelín SAMGs are under internal review and approval. Target for full implementation of SAMGs is the end of 2004.

Evaluation

Transition of AM actions from the EOPs to SAMGs and related responsibilities are well defined with no concurrent use of both packages. Basic criteria for the transition are consistent with the WOG generic concept.

Dedicated human resources for the implementation of SAMGs during an accident are defined. However, the current staffing level of the SAMG evaluation group appears to be rather small (see Section 2.4).

It seems that the intention, within a SAMG strategy, of an intentional release of radioactivity from the containment (as during venting to prevent severe containment challenge) is not clearly defined and understood within the plant organisation. This issue should be resolved when finalizing the ERP and respective plant procedures.

Future actions by the operator/regulator, which are recommended on the technical level to be monitored jointly in the framework of the pertinent bilateral Agreement between Austria and the Czech Republic:

- The plans of the Temelín plant to implement filtered venting as part of SCG-1 should be clarified. Our understanding is that this is a planned measure for implementation as needed (Severe Challenge Guideline SCG-2, “Depressurize Containment”). Confirmation of this could be a subject for further monitoring.

2.4 Organisational Changes at the Plant Relating to SAMP Implementation

VLI No.	VLI title / description
1.4.1	What organisational changes have been made in relation to SAM? Are there any organizational changes other than establishing the SAMG Evaluation Group?
1.4.2	What is the staffing and qualifications of the SAMG Evaluation Group within the TSC?
1.4.3	Have any administrative arrangements been made for the provision of required information to the SAMG Evaluation Group during a severe accident?
1.4.4	Have any additional communication lines needed for the execution of SAMGs during a severe accident been established?

State-of-the-art requirements and practices

Additions to plant's emergency plan and related procedures (as discussed in Sections 2.2 and 2.3) have to be made in relation to SAMG implementation. It should be noted that these modifications are usually not numerous, since the emergency organisation is almost the same in the case of a DBA and in a SA. Most of these changes are implemented based on the existing organisational framework.

Establishing the SAMG Evaluation Group (EG) is not a new process, but is rather evolution of the organisational process which is in place at the plant in relation to DBA conditions. SAMG EG should include adequate number of qualified staff. SAMG EG staff, in particular the decision maker, should have qualifications and experience in control room operation (see Section 3.6.5).

Current plant status

Relevant changes in the existing administrative framework for emergency response are being introduced. The updated ERP and SAMG package are under review and approval. Following their approval the respective emergency implementing procedures will also be updated. The existing EOPs will also be revised to introduce proper instructions for the transition to SAMGs.

To provide appropriate support for the implementation of SAMGs during a severe accident the SAMG Evaluation Group has been established within the existing structure of the TSC. It should be noted that the TSC was established earlier as an integral part of the implementation of symptom based EOPs.

In accordance with the current organisation of the TSC at Temelín NPP [Sy̌kora 03 a], the team responsible for the evaluation of SAMGs during a SA (SAMG EG) includes 4 persons, namely: the Shift Supervisor, Safety Engineer, Operational Support Engineer, and Intervention Control Engineer (it has been pointed out that the latter one is the person who has very good expertise in line-up of equipment and will be capable to advise on non- conventional line-up). They are supported by Radiation Protection, Radiochemistry Support and Information Systems specialists.

Evaluation

Changes in the administrative framework including ERP, EOPs, and administrative procedures related to SAMGs are specified and their implementation ongoing. The current status is consistent with the actual stage of the SAM implementation process. Some concerns related to the staffing of SAMG EG are discussed below.

Based on practices adopted in other plants, the SAMG evaluation group should include 3-4 persons who are responsible for the evaluation of plant conditions and SA challenges and for the determination of SAM strategies using SAMGs. This group should be dedicated to SAMG evaluation and not have any other responsibilities. The Operational Support Engineer and Intervention Control Engineer (or their equivalents) are part of the TSC anyway. During a SA they may have other duties and might not be able to fully concentrate on the evaluation of strategies using SAMGs.

Future actions by the operator/regulator, which are recommended on the technical level to be monitored jointly in the framework of the pertinent bilateral Agreement between Austria and the Czech Republic:

- The Specialist's Team would also recommend the Austrian Governement the consideration to verify in due time that observations from the SAMG validation process on the performance of SAMG EG staff and conclusions regarding the adequacy of staffing and organisation of this group within the TSC are properly collected and, if needed, appropriate organisational changes made.

3 TEMELÍN SAM PROGRAMME EVALUATION

3.1 Accident Analysis done by CEZ-ETE to Support SAMG Programme

Severe accident analysis that is intended to support the development of SAM and SAMGs should fulfil multiple objectives. They include understanding of major SA phenomena including their timing, gathering insights relating to plant behaviour during SA including source terms, identification of plant weaknesses to SA, identification of SAM mitigative measures/actions and related plant improvements, and validation of SAM measures with respect to their effectiveness and consistency with plant design capabilities.

It should be noted that a considerable number of deterministic severe accident analyses is normally performed to support the development of Level 2 PSA. In addition, the plant specific PSA itself provides useful insights relating to the risk.

Supporting accident analysis used to backup the development and implementation of AMP can be evaluated based on good practice criteria described in the IAEA guidelines [IAEA 03, IAEA 99]. These criteria address both the quality of the analysis and its scope/ level of detail. The following sections address these aspects in more detail.

The availability and use of plant specific PSA is discussed in Section 3.1.1 Section 3.1.2 provides some background information related to the scope and level of detail of the existing severe accident analysis. Modelling aspects of severe accident analysis are discussed in Section 3.1.3, documentation of the analyses in Section 3.1.4, and the related quality assurance (QA) aspects in Section 3.1.5.

3.1.1 Availability of PSA and Other Supporting Safety Studies

VLI No.	VLI title / description
2.1.1	Has the plant specific PSA been completed? Is there any work for PSA update on-going?
2.1.2	Were the plant specific PSA results available for the identification of SAM strategies? How these results have been used in the development of SAMP?
2.1.3	Have Level 1 accident sequences beyond 24 hours been investigated in relation to SA?
2.1.4	Have any plant specific severe accident insights/strategies been identified in the PSA study?

State-of-the-art requirements and practices

A plant-specific PSA and supporting accident analysis are recognised as very important elements in the development of SAM programme and SAMGs. They serve as an important means to ensure that the SAM guidance prepared is appropriate for the plant, in terms of identifying potential challenges, verifying applicability of strategies, and supporting implementation activities such as guideline validation. One of the first tasks in developing the plant specific SAMP is to identify the severe accident phenomena and scenarios, which can result in failure of the plant fission product boundaries. PSA, if available, is the best information source for these investigations.

A comprehensive PSA Level 1 and 2 provides a tool, which can be used to identify plant vulnerabilities and also to measure the effectiveness of accident management measures to reduce risk. Preventive AM measures can be incorporated into level 1 PSA, and the mitigative measures modelled in the level 2 PSA. The reduction in predicted risk profile (in terms of the source terms and their frequencies) is a measure of the impact of implementing AM. It should be noted that both Level 1 and Level 2 parts of the PSA are needed in order to be able to take credit for both preventive and mitigative measures.

In the USA practice [NRC 88, NUMARC 92-01, NEI 94], the issues related to severe accidents have been addressed through the assessment of plant specific Individual Plant Examinations (IPEs) and other supporting studies. This assessment focused on providing information on relevant fission product release pathways, contributing severe accident phenomena, dominant core damage sequences and contributing failures. It includes any 'insights' identified during the performance of the IPE, which would reduce risk and any 'vulnerabilities' reported in the IPE in response to Generic Letter 88-20 [NRC 88, NRC 90].

So far not many plants of WWER 1000 type have Level 2 PSA. A preliminary PSA study has been developed for Balakovo NPP (WWER 1000/320). It can be noted that in this study containment bypass scenarios (mainly PRISE) have been found to be a dominant contributor to the radiological risk of the plant [Morozov 03].

Current plant status

The Temelín specific PSA (Level 1 and 2) was developed in 1996. This study is currently being updated in order to reflect the actual knowledge of plant design and procedures (symptom based EOPs introduced in 1996), state-of-the-art PSA techniques, resolution of IPERS recommendations to the original 1996 study, the final plant design, and experience gained from other PSAs conducted for WWER 1000/V320 units.

Summary results of Level 1 updated study (2001 status) are provided in Refs [ČEZ 02, Mlady 03 a, Mladý 03 b]. PSA Level 2 upgrading study is reported being underway (in co-operation with Scientech Inc.). The ongoing project on updating the Level 2 PSA is to be finalized by the end of 2003. Preliminary results from this study have been made available for the PN 7 project during the Prague workshop [Mlady 03 a, Mladý 03 b].

The Temelín PSA is reported to be a full scope study, which addresses internal and external initiating events and covers full power operational states and shutdown conditions. However, risk contributions from external events and shutdown risk were not discussed in details in the presentations given during the Prague workshop [Mlady 03 a]. It seems that this part of the PSA upgrading study is not completed yet. The upgraded PSA is reported to incorporate further improvements in modelling (e.g. common cause failures, treatment of dependencies, more realistic success criteria, etc.) and data (updated using WWER operating experience). The PSA includes uncertainty analysis addressing all the modelling parameters included in the PSA database. No information was provided on the sensitivity analysis with regard to modelling assumptions.

It was reported that the most significant SA scenarios to be used in the development of SAM and SAMGs were identified in 1996 based on PSA Level 1 and 2 [Sýkora 01 a]. These included three basic groups of severe accidents scenarios: the primary to secondary LOCA (PRISE) without operator actions, large RCS LOCA without emergency core cooling system (ECCS), and loss of off-site power (LOSP) leading to station blackout.

At that time the core damage frequency (CDF) was clearly dominated by the scenarios initiated by PRISE, (73% of the total CDF) followed by large LOCA (8,8%), small LOCA (4,5%) and LOSP (3,0%). In the most recent PSA Level 1 [Mlady 03 a], this risk profile changed considerably. The new CDF risk profile is relatively well balanced such that there are no dominant initiating events for accidents or individual combinations of failures (minimal cutsets) leading to core damage [Mlady 03 b]. The dominant scenarios identified in the old study remain the risk significant contributors (small LOCA – 22,1%, medium PRISE – 20,7%, LOSP – 17,8%) in the updated internal events analysis.

Evaluation

A plant specific PSA was available and used in the identification of plant specific vulnerabilities to SA and development of AM strategies. Input from the upgraded PSA is being used in finalizing implementation of SAMGs. However, the full scope Level 2 PSA has not been completed yet (risk related to external events and shutdown risk is not considered).

It should be noted that external events may have an impact on the risk-profile however preliminary results presented at the Prague Workshop indicate that the core damage frequency impact is not expected to be large and these results are not expected to bring insights on any new SA challenges. Therefore, this is not an important issue from the point of view of SAMG development. Shutdown PSA may generate some new issues that should be addressed within SAM. So far, only very few plants (Borssele, Netherlands; Goesgen, Switzerland; and Koeberg, South Africa) addressed the accident scenarios initiated during shutdown conditions in plant specific SAMGs. For Temelín, this issue can be addressed after SAMGs for full power operational states are implemented. Based on the experience from other plants, extending full power SAMGs to cover shutdown conditions is relatively simple [Van Haesendonck 01].

Based on the results presented so far, it can be concluded that in terms of core damage (CDF) the current risk profile of the plant is relatively well balanced. It is worth noting that although the relative contributions to CDF changed as compared to results obtained in 1996 PSA study, the scenarios selected in 1996 as the basis for the development of SAM are still valid.

The information presented during the Prague workshop [Mlady 03 a] seems to indicate that the plant specific PSA reflects the current state-of-the-art in the area of PSA. The original 1996 study was subject to external review (IAEA IPERS mission) and the resulted recommendations are reported incorporated in the upgraded study. Detailed evaluation of the PSA study within the PN7 project was beyond the scope. However, it is believed that the study was performed in compliance with the current state-of-the-art. Some observations can be made with regard to PSA modelling based on the presentations given during the Prague workshop. They are briefly discussed below.

The initiating event frequency used in the PSA model for small break LOCA is very conservative as compared with the available world--wide industry data. This issue is not expected to affect the selection of SAM strategies, but it certianly has an impact on the plant risk profile and perhaps on decisions on the allocation of resources to reduce SA risk.

It was shown that the Level 1 model (CDF) had been subject to uncertainty analysis with regard to reliability parameters [Mlady 03 a]. Sensitivity analyses on basic modeling and phenomenology assumptions were not discussed at the Prague meeting. This issue seems to be important in relation to Level 2 PSA (indeed it is important in all Level 2 PSA studiesw) which is subject to considerable uncertainties. For instance, the impact of early hydrogen burn on accident progression was not investigated. It is worth noting that without dedicated devices for deliberate ignition, which seems to be the case for Temelín, the assumption that hydrogen deflagrations occur at the early phase of accident involve considerable uncertainty (see related comment in Section 3.1.3). Therefore, the risk impact of this assumption should be assessed by the sensitivity analysis.

Some inconsistencies between the PSA and the existing supporting analyses can be observed. For instance, there is a relatively high contribution (~ 28%) of 'early containment failures' due to basemat melt (through the penetrations and instrumentation channels located in the cavity wall) [Mlady 03 a], which seems to be inconsistent with the results obtained from severe accident analyses [Kujal 03, pages 16-18]. These SA calculations indicate that the rate of corium-concrete interaction in the region of vertical gauge channels is by one order of magnitude lower than in the reactor cavity region. Therefore, a failure of basemat by melting through the vertical gauge channels should be expected later than failure due to basemat-melt in the cavity region. This failure is hard to be considered as an ‘early’ failure. This issue does not have any impact on the selection of SAM strategies, however, there appears to be an incoherence between the deterministic severe accident calculations and the probabilistic modeling of the phenomena in the PSA. This issue may indicate deficiencies in QA when carrying out safety analyses.

It seems that the interface between the PSA team and thermal hydraulic (T/H) accident analysis team needs improvement. Basic QA rules applicable to safety analysis should be followed also in the PSA area. General comments on QA programme in safety analysis are also given in Section 3.1.5.

The existing Level 2 PSA should be used to assess the effectiveness of ‘mitigative’ AM measures proposed within the plant specific SAMGs. The Level 2 PSA model is indeed capable of quantifying the fission product release frequency and magnitude and should be used to assess the impact of SAM in terms of the source term. No source term results, which could support such evaluation, were reported for Temelín at the Prague Workshop.

3.1.2 Scope and Level of Detail of the Existing Severe Accident Analysis

VLI No.	VLI title / description
2.2.1	Is the level of detail of accident analysis adequate to support all needs of SAMP development? How many accident scenarios have been analysed for accidents scenarios without operator’s actions and for scenarios with AM measures?
2.2.2	What scenarios have been analysed with regard to system success criteria (e.g. timing and rate of ECCS injection during the in-vessel or ex-vessel phase)
2.2.3	What potential plant upgrades in relation to SAM has been identified as the result of severe accident analysis?
2.2.4	Certain Temelín severe accident progression calculations were reported in the Melk Protocol meetings in April and September 2001, and in the CEZ-ETE presentation at the NEA-OECD meeting in September 2001. What, if any, additional severe accident scenarios have been modelled in connection with SAMG development and implementation?

State-of-the-art requirements and practices

Severe accident scenarios that should be selected for the analysis to support SAM can be grouped into two major categories: accidents without operator actions/AM measures and accidents in which AM measures are considered.

The first group includes sequences that would lead to core damage, core melt, vessel failure, containment failure, and release of fission product to the environment. Accidents of this type are analyzed as a first step of SAM investigations to gain an understanding of the plant behaviour during SAs, to determine severe accident phenomena important for the specific design, and to understand and rank challenges to fission product barriers.

Since the number of sequences leading to the release of radioactivity is very large the selection is made based on a systematic categorization of sequences. The categorisation scheme is typically based on several state attributes such as initiating event (IE) group and status of safety systems (emergency core cooling, secondary heat sink, containment heat removal and containment boundary). Some guidance can be found in the IAEA guidelines [IAEA 03]. The number of sequences within this group depends on the applied grouping of IEs and in PWR practice is of the range 20 – 30 sequences. For instance, SA simulations conducted at the initial stage of SAM investigations (for the WWER 440/213 plant) within the Phare project 4.2.7.a and 4.2.7.a/93 [WESE 96, WESE 97] included 20 SA sequences. It should be noted that insights from plant specific PSA provide a sound basis for the selection of most important scenarios (as discussed in Section 2.1).

At the later stages of SAM investigations the analyses focus on confirmation and optimization of SAM strategies. Typically, scenarios selected in the initial phase are revisited and recalculated assuming specific operators actions and AM measures. Focus is put on system capabilities to perform the required functions, selection of symptoms to be used to initiate AM measures, investigation of time margins for AM actions, and identification of potential plant upgrades. Additional input is also required for the calculation of setpoints used in SAMGs and development of computational aids.

Current plant status

The analytical work to provide support for the development of Temelín SAM was initiated in 1991. [Sy̆kora 01 a]. The analyses of SA scenarios were performed by Nuclear Research Institute (ÚJV Řež) which acts as TSO for all Czech NPPs.

In the period of 1991-93 a number of SA scenarios initiated by various LOCA events were simulated using the STCP-M package (MARCH3-M code). These scenarios included both high and low pressure scenarios. A broad spectrum of LOCAs (25, 40, 100 and 2×850 mm) and transients (plant blackout, loss of FW to SGs) were addressed in these calculations. Scenarios that involved consequential LOCA events (caused by a station blackout and loss of RCP seal cooling) were also addressed. Source Term calculations were performed for selected cases which had high frequency of occurrence or involved early and severe radiological consequences.

The calculations focused on the investigation of accident scenarios without operator actions. Several cases, which differed with the availability of safety systems (ECCS, CSS, EFW, electrical power, etc.), were simulated for each of the accident initiators. Investigations of AM measures were limited.

Specific SA scenarios covered in these calculations are described in the paper presented by Kujal [Kujal 94]. They include 19 simulations – 2 cases for LOCA 25 mm, 6 cases for LOCA 40 mm, 1 case for LOCA 100 mm, 4 cases for LB LOCA (2×850 mm) and 6 cases for transient group). One of the LOCA 40 mm cases was intended to address impact of corium spreading outside the reactor cavity.

The calculations provided early insights on plant vulnerabilities and potential AM measures (RCS depressurization and opening of the reactor cavity to slow down progression of molten core concrete interaction (MCCI)).

Starting from 1994, analytical investigations of SAs were continued using more advanced simulation codes, improved plant and component models, and consistent input data. All activities were subject to a systematic QA. These simulations were performed using the MELCOR code version 1.8.3 (adopted for WWER 1000 NPPs), ICARE-2 (analyses of fuel damage) and CONTAIN code version 1.12 (analyses of containment phenomena).

An overview of the calculations performed with the advanced SA codes is given in the material provided by Czech side or published elsewhere [Sy̆kora 01, SONS 01, Pazdera 03, Kujal 03].

Two scenarios – medium break LOCA 100 mm with station blackout and transient without operator actions – were calculated in the initial phase of these investigations [Sy̆kora 01 a]. The latter scenario was analysed as a typical high pressure SA scenario accompanied by SA phenomena (DCH, hydrogen burn and basemat melting through).

New calculation cases were defined and investigated when PSA results become available. These included three groups of scenarios: Large primary to secondary (LOCA (40 mm) without operator actions (PRISE), LB LOCA (DN 200 mm) on pressurizer surge line (without active ECCS, and station blackout. There were specific cases within the PRISE and LB LOCA groups that addressed impact of selected SAM measures (corium spreading over the expanded cavity area) or certain SA phenomena (direct containment heating - DCH), steam explosions, hydrogen explosion, long term overpressurization of the containment). One of the recent calculations investigated the scenario in which LP ECCS is restored in the ex-vessel phase [Pazdera 01].

More detailed description of the scenarios can be found in Annex A. The number of calculated cases performed in this phase of the investigations is not known. It was reported that further analyses are planned be performed in the end of 2003 using the latest version of MELCOR code (1.8.5).

The calculations provided insights to support decisions on the implementation of various SAM strategies and potential plant upgrades. The relevant insights obtained from the analyses are discussed below.

The penetration of the basemat by molten corium was shown to be a significant threat to the containment integrity. Corium spreading to the room adjacent to the reactor cavity has been found to reduce the rate of molten corium concrete interaction (MCCI) during ex-vessel phase of a SA. The corium spreading strategy was therefore adopted at Temelín. Related planned modifications of the plant include the installation of a remote control system to open the reactor cavity door before the RPV break and placing removable physical barriers to restrict molten corium pool area and protect equipment hatch against MCCI. Additionally, the existing penetrations and reactor cavity instrumentation channels are planned to be protected for the case of molten corium attack by individual concrete plugs.

The protection against long-term overpressurization of the containment needs to be provided. The existing containment pressure test depressurization line with filters and throttle valves as the ultimate means of containment venting is proposed to be used to implement this strategy.

Evaluation

The extent of SA analyses performed for Temelín (in terms of a number of SA scenarios without the operator actions covered) is sufficient to provide a good understanding of most SA phenomena and challenges to the plant. From this point of view the situation is similar to other plants. Insights obtained from the analysis have been used to support SAM strategies.

However, large part of these simulations is old and they do not necessarily reflect the current plant status and state-of-the-art in the area of SA codes and modelling. Newer calculations performed with MELCOR code version 1.8.3 are limited to several cases.

It seems that no calculations were performed recently to assist in the optimization of SAMG strategies and with regard to system operation success criteria (e.g. timing and rate of ECCS injection during the in-vessel or ex-vessel phase). The situation should improve after completion of the set of analyses planned for the end of 2003 (using MELCOR version 1.8.5).

3.1.3 Modelling Aspects of the Existing Severe Accident Analysis

VLI No.	VLI title / description
2.3.1	What computer codes were used in the severe accident analysis?
2.3.2	Which organisations were involved in the calculations of severe accidents (external subcontractors or plant staff)?
2.3.3	What qualifications have the personnel involved in the calculation?
2.3.4	To what extent the SA analysis is based on best estimate assumptions?
2.3.5	Were all phenomenological criteria/assumptions identified and clearly defined for the following issues: (a) Fuel cladding and fuel degradation/failure, (b) Zr-UO_2 and Zr-steel reactions, (c) Core relocation, (d) Failure of internal supports and RPV bottom head, consideration of creep failure mechanism, (e) Direct containment heating, (f) Debris transport, quench and coolability of corium, (g) Steam explosion in RV and cavity, (h) Accident induced rupture of the RCS piping and SG tubes, (i) Accident induced loss of containment.
2.3.6	Have benchmarks been conducted for verification of models and data used in SA analysis?
2.3.7	Have the models been verified by code to code comparison tests, sensitivity studies, and comparison with the results obtained for other similar plants? Which scenarios were subject to such comparison? What were the conclusions from such comparison?
2.3.8	Have modelling assumptions and/or parameters been subjected to sensitivity/-uncertainty analysis? Which parameters were subject to this analysis? Which accident scenarios were selected for these investigations?
2.3.9	Have benchmarks been conducted for verification of models and data used in SA analysis?
2.3.10	Have the models been verified by code-to-code comparison tests, sensitivity studies, and comparison with the results obtained for other similar plants? Which scenarios were subject to such comparison? What were the conclusions from such comparison?
2.3.11	Have modelling assumptions and/or parameters been subjected to sensitivity/ uncertainty analysis? Which accident scenarios were selected for these investigations?

State-of-the-art requirements and practices

Beyond design basis accidents and severe accidents are analysed with ‘realistic’, best estimate codes. These codes include integrated codes and mechanistic separate effects codes.

Integrated codes combine models in one package: for heat transfer, fluid flow, fission products release and transport, plant system operation and performance, and operator actions. The codes of this type incorporate physical models for processes that are important during transients leading to and go beyond fuel damage, and all models are coupled at every time step. The best known and most frequently codes of this type are MELCOR and MAAP.

MELCOR has been developed by Sandia National Laboratories for the US Nuclear Regulatory Commission (NRC) and succeeded the existing Source Term Code Package [SNL 91]. This code is most frequently used for severe accident analysis in WWER NPPs.

MELCOR is a fully integrated, relatively fast-running code with the flexibility to model a wide spectrum of severe accident phenomena in light water reactor nuclear power plants. Characteristics of severe accident progression that can be treated with MELCOR include the thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup and degradation; radionuclide release and transport; hydrogen production, transport, and combustion; core-concrete attack; heat structure response; and the impact of engineered safety features on thermal-hydraulic and radioactive effluents behaviour. The use of parametric models is limited in general to areas with great uncertainties where there is no consensus concerning an acceptable mechanistic approach.

MAAP is a fully integrated code that can simulate the response of light water reactor plants during SA including actions taken as part of the accident management. The spectrum of severe accident phenomena covered in MAAP 4 is similar to the one represented in MELCOR. The code uses a control volume and flow approach, however, the geometry of the control volumes is pre-specified (depending on reactor type).

Mechanistic codes are applied for detailed investigations of individual phases of the accident progression, to model separate effects and /or where it is felt that model details in integrated codes is not sufficient to produce reasonable results. Examples for widely used mechanistic codes are SCDAP/RELAP5, CATHAR-ICARE and ATHLET-SA (in-vessel phenomenology) and COCOSYS (RALOC), CONTAIN (for containment studies including hydrogen and FP behaviour)

A key element in using codes (both mechanistic and integrated) is the demonstration that the codes' predictions may be fully relied upon when providing the basis for accident management programmes to be established. This needs to be done by qualifying the code against experimental evidence, other codes, and engineering judgement.

The recognized computer codes that are applied for SA simulation are normally subject to extensive verification and validation (V&V). V&V benchmarking covers a range of phenomenological issues. Some validation efforts were also undertaken on larger integral standard problems.

MELCOR has been successfully validated by the code developers through multiple SA standard problems and code-to-code comparisons [Gauntt 01]. The results of code-to-data comparisons indicate that with the proper choice of user-specified modeling parameters, the code can match the general trends observed in the experiments and predicted by the more detailed codes (such as SCADAP/RELAP5). However, a relatively strong 'user's effect' can be observed.

Results of code-to-code comparisons have shown a wide divergence in the predicted responses of the plants. The highest uncertainties that may have impact on SAM measures and containment challenges exist in the areas of core loss of geometry, in-vessel and ex-vessel corium-water interaction, hydrogen combustion, and molten corium concrete interaction (MCCI).

It should be noted that validation of the currently used codes was carried out on experiments that do not necessarily reflect WWER plant specific features. This is an additional contributor to the uncertainties associated with the results of SA obtained for Temelín NPP, and especially affects the in-vessel portion of the accident progression calculation due to differences in reactor coolant system layout (e.g., horizontal steam generators) and reactor pressure vessel internals design, and the ex-vessel portion of the analyses insofar as differences in containment layout are concerned.

In some cases the existing uncertainties may have an impact on the selection of specific SAM measures. Therefore, in the plant specific applications, due attention should be given to the assessment of uncertainties using code to code comparison and sensitivity analysis with respect to various modeling assumptions and parameters. The sensitivity/benchmarking tests should be accompanied by the engineering evaluation.

The plant specific analyses conducted for the development of AM should be based on best a estimate approach and using evaluation models for the plant. This involves selection of best estimate correlations and use of realistic assumptions and criteria. One of the relevant aspects is the selection of suitable core damage criterion. This criterion should be 'realistic' and also consistent with the approach taken in SAM. The criterion for the onset of core damage recommended by Westinghouse for the use in AM is based on a directly measurable parameter. The parameter used in the Westinghouse EOP/SAMG approach is the reactor coolant temperature at the core outlet measured by the coolant thermocouple system. The value of 650 ^{o}C is recommended by Westinghouse to be used as an indication of the onset of core damage.

It is important that all relevant assumptions, criteria and input parameters that may affect the results be identified and clearly defined. Attention should be given to the assessment of applicability range of the models and correlations and their suitability with regard to plant specific features. In the computer codes used currently for severe accidents simulation there are a number of modelling options and the need for user definition of model parameters.

Specific modelling issues that involve analyst decisions such as in-vessel core degradation phenomena, consideration of creep failure mechanism, direct containment heating (DCH), debris transport, quench, and coolability of corium, steam explosion in RPV and cavity, accident induced rupture of the RCS piping and SG tubes, and the accident induced loss of containment should be subject to comprehensive engineering evaluation. The evaluation should include the sensitivity analyses.

Current plant status

Analytical tools applied in SA analysis at the Temelín plant have been described in the presentations [Duspiva 03, Jakab 03]. The tools include a number of integral and mechanistic separate effect codes.

The main computer code used in the severe accident analysis for Temelín was MELCOR 1.8.3. The Czech regulatory authority (SUJB) has certified this version. The newest version of the MELCOR code (1.8.5) is under adaptation (the existing analyses are planned to be updated using this version of the code).

In addition to MELCOR integral code, some mechanistic separate effects codes have also been used – ICARE-2 for the analyses of fuel damage progression during SAs and CONTAIN code (version 1.12) for the analyses of containment phenomena including DCH aspects. The code WECHSL, which is part of the integral code ASTEC being developed in EU, was used for MCCI calculations.

The fission product releases were calculated with the mechanistic code IODE, developed in Switzerland and successfully tested in International Standard Problem ISP-41 [Duspiva 03].

The Institute for Nuclear Research (ÚJV) Řež has conducted all SA analyses for Temelín NPP; it works at the same time as a TSO for Czech NPPs and SUJB. ÚJV Řež was deeply involved in the severe accident analysis from the early 1990s. The personnel involved in the calculation are highly qualified and have necessary expertise including the knowledge of severe accident phenomenology, familiarity with the tools and detailed knowledge of the plant.

This team is reported [Duspiva 03] to actively participate in the international research activities (CSARP, EU FWP, OECD) and code validation exercises. In relation to MELCOR, the expertise was consolidated through considerable support from MELCOR developer team (MCAP) and international exchange of user experience (MCAP, EU FWP, ISP).

Evaluation

From the presented material [Duspiva 03, Jakab 03] it can be concluded that the Czech experts involved in the SA analysis for Temelín NPP have access to relevant computer codes, information and data, know-how, and lessons from practical experience. They are deeply involved in international programmes in the area of SA research, development and validation of analytical tools, and their application. Several examples where given during the Prague workshop that support this statement.

Efforts to reduce considerable uncertainties in the results of SA analysis, which is one of the generic issues, are clearly visible. The team tries to use the state-of-the-art codes and experience from other code users. However, some concerns can be expressed in relation to the uncertainty and sensitivity analyses. No information was provided on this subject in the workshop presentations. The highest uncertainties, which may have an impact on SAM measures and containment challenges (e.g. in the areas of core loss of geometry, in-vessel and ex-vessel core coolant interaction, and hydrogen combustion) should be subject to engineering evaluation and sensitivity analysis.

It should be noted that proper evaluation of the existing Temelín analyses with regard to modelling aspects would require detailed investigations that involve specialized expertise and significant effort. Such evaluation is beyond the scope of this project. In this situation, an independent review of SA analysis conducted or used in the development of SAM (e.g. by IAEA under the IPSART or RAMP missions) would be desirable.

Future actions by the operator/regulator, which are recommended on the technical level to be monitored jointly in the framework of the pertinent bilateral Agreement between Austria and the Czech Republic:

- The Specialist's Team would recommend the Austrian Governement to consider supporting that an independent review is performed for the area of SA analysis. It is worth noting that IAEA RAMP provides such assistance at various stages of SAMP development. RAMP review limited to the area of SA analyses could be conducted before final implementation of SAMG. Such review would also assess the scope of the analyses and quality aspects including those discussed in Sections 3.1.4 and 3.1.5.

3.1.4 Adequacy and Completeness of Documentation of the Accident Analysis

VLI No.	VLI title / description
2.4.1	Are there a plant specific data compiled into a single document/file ("Database of the Analysis")
2.4.2	Is there a comprehensive description on how plant data were converted into a code input deck for SA analysis ("Engineering Handbook")?
2.4.3	Are the input data, assumptions, and model information complete and adequately documented? Is there a complete list of all input variables required for severe analyses code?
2.4.4	Has the selection of optional code/model parameters been properly documented (including justification)?
2.4.5	How have the input data for individual accident scenarios been archived?
2.4.6	Are the results of accident analysis adequately documented? Can the results be properly linked with the specific input deck? Is there a comprehensive description of the accident scenario and the code version used in the simulation?

State-of-the-art requirements and practices

Complete and adequate documentation of the information used in the analysis and results of analyses is important to facilitate proper use of data and to ensure the traceability of results.

All input data, assumptions, and model information should be compiled into a single document ("Database of the analysis") [IAEA 03]. The scope of data is typical for the majority of the integral codes applied in the SA analysis. Data include hydrodynamic component data, physical and chemical data for structures, data for peripheral systems, reactor protection and reactor kinetics data. The geometrical characteristics of the plant and plant performance data should represent 'as built' conditions. All data sources should be referenced. It is recommended that in addition to the database document there is also a comprehensive description on how plant data were converted into a code input deck (so called "Engineering Handbook"). This document should describe relevant features of the model and assumptions.

The collection of data and the preparation of database documentation should be carried out with adherence to a thorough QA procedure. Documentation of the input data should be subject to independent review. Records of the review should be included.

The documentation of the results of accident analysis should include a complete description of the accident scenario, relevant information on the code version, and reference to the specific input deck. All relevant data needed for reproducing the analysis should be available including code modelling options and parameters. Presentation of the results typically includes comprehensive description of plant response (behaviour primary and secondary system, and containment), timing of main events, summary of peak quantities, information on source term, and discussion/interpretation of results. Selected parameters should be represented in a graphical form. Archiving the input deck in electronic form for a reasonable period of time is considered a good practice.

Current plant status

Plant specific data have been collected and are available for severe accident analyses conducted by UJV Rez [Kujal 03].

It is reported that a 'Temelín Database' has been prepared for the area of SA analysis [Duspiva 03]. This database covers all important process systems and safety systems as well as the containment buildings. Plant specific models have been prepared for the core, RCS, secondary system, safety systems, plant control systems and relevant operational systems. No detailed information has been made available on the related documentation, its completeness and adequacy.

Information on the documentation of results of the existing SA anlysis is limited. Some high-level summary information on this subject was provided during the Prague Workshop and in previous meetings in 2001.

Evaluation

Insufficient information is available upon which to draw a conclusion on this aspect of the SAM programme. A Temelín database has been prepared. However, it was not discussed in any detail and was not available for review. None of the severe accident calculations (along with their parameter and modeling choices) was available for review.

3.1.5 QA Aspects for SA Analysis

VLI No.	VLI title / description
2.5.1	Is there a formal QA programme applicable for SA analysis?
2.5.2	What arrangements are in place to ensure that the analyses are traceable and reviewable?
2.5.3	Have any independent reviews of SA analyses been conducted at the plant?

State-of-the-art requirements and practices

Basic QA rules applicable to safety analyses in general should also be followed during performance of the plant specific SA analysis. The following recommendations defined in IAEA guidelines [IAEA 03] are applicable:

- Formalised QA programme document should be developed,
- Analyses team should be qualified for the job (qualifications should be documented),
- Responsibilities of all experts involved should be specified,
- All documents needed for a subsequent review should be recorded,
- Only validated and accepted methods should be used,
- Results of analysis should be independently reviewed,
- All differences found in the review should be resolved before the final use of results,
- Complete supporting analysis should be archived for a reasonable period of time.

Current plant status

It is reported [Sŷkora 01 a] that a formal QA programme applicable to safety analysis was established and executed at the plant and TSO for carrying SA analysis starting from 1994. No details are available relating this programme. A QA programme was also reported applied to the PSA. One of the internal procedures related to QA aspects was mentioned in the presentations during the Prague workshop (CR SONS VDS-030 “Internal Procedure on the Assessment of Computer Codes Used for Safety Related Analyses”) [Duspiva 03].

Evaluation

QA was applied for the SAM supporting calculations starting from 1994. No information is available on the details of this programme.

3.2 Development of SAM Strategies

Generally, a SAM strategy is a method that can be used to recover from or mitigate a specific challenge during an accident that involves core damage. Plant specific strategies are normally developed using a systematic process, which starts with the evaluation of high level generic strategies that are applicable to specific type of NPP (e.g. those defined in [WOG 94] for PWRs). The next step is the evaluation of all lower level strategies that can be implemented within each of the high level strategies using plant specific features.

Important aspects relating to the definition of SAM strategies are discussed in the following subsections. Section 3.1 addresses the selection and definition of strategies. The review of plant capabilities as designed that can be used in the implementation of the strategies is discussed in Section 3.2. Use of external equipment is addressed in Section 3.3. Formal definition and documentation aspects are addressed in Section 3.4.

3.2.1 Selection of SAM Strategies

VLI No.	VLI title / description
3.1.1	Do the selected AM strategies cover all relevant functions? Does it include the subcriticality, restoration of core cooling, protection of RCS integrity, treatment of combustible gases, protection of containment integrity and minimization of radioactivity release?
3.1.2	Do the selected strategies reflect the current international knowledge and practices?
3.1.3	Are there any specific generic (WOG) strategies that have not been considered in the plant-specific SAM? If so, what is the justification for such decisions?
3.1.4	Are the objectives and criteria clearly defined for each of the strategies considered in the plant-specific SAM?
3.1.5	What are the qualitative and quantitative goals of the severe accident management strategies implemented at Temelín?

State-of-the-art requirements and practices

Selection of strategies and their descriptions is the first step in the development of SAMGs. High-level WOG generic AM strategies cover all relevant functions (subcriticality, restoration of core cooling, protection of RCS integrity, treatment of combustible gases, protection of containment integrity, minimization of radioactivity release).

In the majority of the plants these high level strategies have been found applicable. Differences may be observed in the lower level strategies reflecting the availability of plant specific equipment and related instrumentation in a particular accident sequence. Specific plant features may also affect the priorities given to certain strategies versus other.

Current plant status

NPP Temelín adopted the WOG SAMG approach. The WOG high level strategies have been selected based on the available information including EPRI SAMG Technical Basis report which defines 13 Candidate High level Actions, NRC reports: NUREG/CR 5474, -5780, -5856, IPE insights from several WOG utilities, Swedish BERGs, other plant PSAs and severe accident analyses, and plant specific information.

The high level strategies described in the WOG material [WOG 01] are applicable to Temelín NPP. The only procedure that is omitted in Temelín in comparison with the standard WOG SAMGs is SAG-8 “Flood Containment”, which is simply impossible to fulfil in view of the specific design of WWER 1000 containment. The lower level strategies have not been discussed in detail during the workshop. Information was limited to the specification of equipment that is to be applied in the implementation of strategies.

There are 7 Severe Accident Guidelines (SAG) that are entered first at onset of core damage and 4 Severe Challenge Guidelines (SCG) that are entered when there are severe challenges to the containment integrity. The following strategies have been defined for Temelín NPP:

- SAG-1 – Inject into the Steam Generators
- SAG-2 – Depressurize the RCS
- SAG-3 – Inject into the RCS
- SAG-4 – Inject into the containment
- SAG-5 – Reduce Fission product Releases
- SAG-6 – Control containment conditions
- SAG-7 – Reduce containment hydrogen
- SCG-1 – Mitigate Fission Product Releases
- SCG-2 – Depressurize Containment
- SCG-3 – Control Hydrogen Flammability
- SCG-4 – Control Containment Vacuum.

Evaluation

The SAM strategies developed for Temelín are based on Westinghouse Owner Group (WOG) SAMGs, developed in close cooperation with Westinghouse, taking into account the specific features of the Temelín NPP. As the WOG SAMGs are well established and implemented successfully in many plants, the situation in Temelín is considered to be fully satisfactory in this respect.

3.2.2 Adequacy and Completeness of the Review of Plant Design Capabilities

VLI No.	VLI title / description
3.2.1	Has systematic analysis been performed for identification of alternate equipment able to accomplish SAM related functions (pressure and flowrate for pumps, available water sources, power supply, etc.)?
3.2.2	Has systematic analysis been performed regarding the equipment performance outside its design range?
3.2.3	Has the influence of support system failures been properly evaluated (cooling water, power supply, etc.)?
3.2.4	Have potential design modifications of the existing plant equipment been identified in relation to SAM?

State-of-the-art requirements and practices

Important step in the definition of plant specific SA defence-in-depth approach is a review of the plant-specific capabilities to perform basic severe accident recovery strategies (secondary side feed, RCS injection, RCS depressurisation, containment water addition and depressurisation, hydrogen control, etc.). The objective of review is to identify all possible means (systems, structures, components) for achieving safety objectives, even those involving the use of equipment outside its original design envelope. At the same time major equipment limitations (for example shutoff differential pressure for injection systems, maximum achievable flow rates, depressurisation efficiency and effectiveness, etc.) should be identified.

The definition of symptoms, and the associated plant process parameters, which must be monitored in order to detect and rank potential challenges and prioritize them for corrective actions is an important next step. It represents the formulation of the basic objectives of the strategies in terms of the safety functions to be protected or restored, if perturbed (or challenges to be prevented). Strategies must then be developed which provide all practical means to protect the safety functions. During this phase it is important to clearly and unambiguously define specific criteria such as entry and exit conditions, diagnostic symptoms, etc. Efforts should also be made to demonstrate that the list of strategies developed is complete. An appropriate verification and validation programme must check the 'correctness' and 'usability' of the strategies.

Current plant status

A Westinghouse team in close co-operation with the competent plant staff conducted the review of plant specific capabilities. All relevant steps as described above have been followed [Dessars 03, Sỳkora 03 b].

Evaluation

WOG SAMG adaptation to Temelín NPP was similar to adaptation to any Westinghouse designed PWRs. The plant specific capabilities to implement the strategies have been identified and properly considered in the SAMGs. Symptoms that must be monitored and specific criteria used in the guidelines are clearly defined.

A team that has considerable experience in the area of SAMGs performed the adaptation to Temelín NPP.

3.2.3 External Information to Support the Development of SAM Programme

VLI No.	VLI title / description
3.3.1	Have all needs for the use of external equipment and related input been determined?
3.3.2	Was the external input made available for the development of SAM?

State-of-the-art requirements and practices

The identification of possible requirements to bring in the equipment from outside the plant should also be addressed. In some areas such equipment can easily be used increasing the number of alternative AM options and reducing the related risk contribution. Information on the availability of such equipment and related time element as well as the equipment capabilities should be realistically assessed.

Current plant status

No external equipment was explicitly mentioned in the adopted SAM strategies.

Evaluation

The potential for using some equipment from outside the plant for SAM should be addressed in finalizing SAM strategies, and could be the subject of a discussion at a Czech-Austria nuclear issues bilateral meeting.

3.2.4 Formal Definition and Documentation of SAM Strategies

VLI No.	VLI title / description
3.4.1	Is basic information on the strategies completed within the AMP (including entry and exit conditions, equipment/systems used, diagnostic means and tools)?
3.4.2	Have the selected AM strategies been formally documented?
3.4.3	Has the effectiveness of the strategies been proven by specific accident analyses?
3.4.4	Has an independent review of the AM strategies been conducted? Who performed the review and which documents were subject to the review? What are conclusions/recommendations from the review?

State-of-the-art requirements and practices

The selected strategies should be documented. Typically, this document is considered a part of SAMG-related background material. It provides comprehensive description of the strategy, its purpose/objectives, and identification of interfaces with the diagnosis tools and specific guideline (including entry and exit condition).

Plant specific equipment/systems used for achieving safety objectives should be discussed. Basic characteristics of component/system and support systems needed for its operation should be provided. This discussion should include major equipment limitations (for example shutoff heads for injection systems, maximum achievable flow rates, depressurisation capacity, etc.) and survivability of equipment (see Section 4 for further discussion of this aspect). Possible negative impacts of implementing the strategy using available equipment should also be identified.

This background material should also provide information on the relevant plant specific instrumentation used as diagnostic means and tools for the specific strategies. Basic characteristics of instrumentation including the redundancy and measurements range should be provided. Survivability of instrumentation and reliability of the readings in severe accident conditions should also be discussed.

Current plant status

The documentation of Temelín SAMGs has been described in the workshop presentations [Sỳkora 03 b]. It is composed of 6 Volumes. Volume 1 includes all guidelines used at the TSC (DFC, SAGs, SCST, SCGs, SAEGs, and CAs). Volumes 2 and 3 provide background material for DFC and SAGs (Volume 2) and for SCST, SCGs, SAEGs, and CAs (Volume 3). Volume 4 includes general user's guidelines, technical basis for setpoints, description of diagnostics and instrumentation, and description of SAMG strategies. Volume 5 provides information relating to interfaces with E-plan and EOPs, and a description of plant capabilities to be used in SAMGs. Volume 6 includes control room guidelines (SACRGs) with the related background.

Evaluation

The SAMG documentation developed for the Temelín plant staff was found to be suitably structured. The subjects addressed in this documentation are those, which would typically be expected for SAMGs based on the Westinghouse Owners Group approach. Since the SAMG documentation was not made available for review, no assessment of the completeness of the information, the quality of presentation, and the usability of the whole SAMG package are possible.

3.3 Performance of Equipment

Performance of equipment, which is involved in AM, is one of the relevant aspects that should be considered in the development of SAM programme and SAMGs. Related issues are discussed in the following subsections.

Sections 3.3.1 and 3.3.2 address general aspects related to the quality of safety equipment. These general aspects are relevant for any accidents and included here to provide a broader safety perspective. Equipment performance in severe accident conditions is the key issue addressed in Section 3.3.3. Severe accident aspects relating to monitoring of the plant in severe accident conditions and related instrumentation are discussed in Section 3.3.4. Some aspects relating to initiation of equipment operation are addressed in Section 3.3.5. Provisions for preserving plant control capabilities in severe accident conditions are covered in Section 3.3.6.

3.3.1 Required Quality of Equipment Involved in Accident Management

VLI No.	VLI title / description
4.1.1	Is there a systematic program at the plant to select reliable manufacturers?
4.1.2	Were the QA systems, used by equipment manufacturers, subject to verification?
4.1.3	Has the operational QA program at the plant been subject to independent peer review? Who performed the review? Have the recommendations been implemented?

State-of-the-art requirements and practices

A systematic programme should be established at the plant for controlling the procurement process. This is normally covered in the operational QA plant programme. Reference bases for the assessment of plant QA status can be found in the Code and in Safety Guide Q6 of the IAEA Safety Series 50-C/SG-Q [IAEA 88, IAEA 96]. Assessment of the plant status relating to this area is beyond the scope of this project. Only some general QA aspects that are relatively easy to verify can be addressed.

Specific interest should be given to a two-step approach to the procurement of safety related items that includes the qualification of the supplier and the evaluation/approval of any individual purchase of items or services. In this approach the supplier should be in an approved list, having a QA programme compliant to the QA requirements. QA system implemented by equipment manufacturers included in this list should be subject to independent verification. Procurement of safety related items should be arranged using only the qualified suppliers. Any individual purchase should be further evaluated and approved prior to placing a purchase order. The classification of items and services with regard to safety used at the plant is one of the related aspects.

Independent reviews of the operational QA programme at the plant should regularly be conducted. Recommendations from such reviews should be evaluated and appropriate actions implemented.

Current plant status

Design and manufacturing quality and the operational care is specifically mentioned by the plant staff in the Prague workshop presentations as one of the important elements of the 'defense-in-depth' concept implemented at Temelín [Sŷkora 03 a]. However, no detailed information is available regarding the operational QA programme.

Evaluation

No evaluation was possible with regard to this aspect. This aspect should be addressed within an independent review of QA programmes.

3.3.2 Preventive Maintenance and Surveillance of AM-Related Equipment

VLI No.	VLI title / description
4.2.1	Is there a documented plant programme for preventive maintenance and surveillance? Is the implementation of this programme properly supported by plant procedures?
4.2.2	Is the extent and frequency of preventive maintenance optimized with consideration given to safety significance of the equipment (risk informed approach)?
4.2.3	Are the provisions for minimizing human errors during maintenance defined?
4.3.4	Is all the relevant equipment subject to regular functional testing? What are the schedules for testing and/or inspections?
4.3.5	Is the inspectability of safety equipment adequate?

State-of-the-art requirements and practices

Appropriate preventive maintenance and surveillance of safety-related equipment is one of the relevant aspects that have impact on the performance of equipment during accident conditions. The assessment of plant status with regard to this aspect is beyond the scope of this project. Therefoe, only some general aspects that are relatively easy to verify are addressed in the related VLIs.

The existence of a documented plant programme for preventive maintenance and surveillance at the plant is essential [IAEA 03]. This programme should be properly supported by plant procedures. It is considered a good practice where the extent and frequency of preventive maintenance have been optimized with consideration given to safety significance of the equipment (risk informed approach). Provisions for minimizing human errors during maintenance should be defined and addressed in the related procedures.

All the relevant safety equipment should be subject to regular functional testing. The schedules for testing and/or inspections of this equipment should be comparable to the practices of nuclear industry elsewhere. Adequate inspectability of safety equipment is a relevant issue related to this aspect. It should be noted that not all equipment that is credited in AM is classified as safety or safety-related and subject to regular testing/inspection.

Current plant status

General provisions for preventive maintenance and surveillance of safety-related equipment are similar to those implemented in other NPPs in Europe. It is known that Czech utilities are interested in the application of risk informed strategy to optimize the preventive maintenance and testing of safety related equipment. No information is available regarding plant specific situation in this regard.

Evaluation

There is not enough information available on preventive maintenance and surveillance of accident management-related equipment upon which to base an evaluation. The plant licensing processcovers preventive maintenance and surveillance of safety-related equipment, however it is recognized that accident management-related equipment includes additional components and systems beyond those designated as ’safety-related‘.

The Pn7 Specialist Workshop focused more on gaining an understanding of the Temelín accident management strategies (and the underlying accident progression analyses), and obtaining information about plant modifications to support those strategies (modifications which were first identified at the Specialist Workshop). Preventive maintenance and surveillance of accident management-related equipment could therefore be recommended to be a subject upon which to receive a presentation from the appropriate Czech experts at one of the next Czech-Austria nuclear issues bilateral meetings.

3.3.3 Equipment Performance in Severe Accident Conditions

VLI No.	VLI title / description
4.3.1	Have the environmental conditions under severe accidents been determined and documented?
4.3.2	Has the equipment performance under accident conditions been evaluated and verified? Has this process been conducted also for non-safety equipment which is considered in SAMGs?
4.3.3	Have 'as-built' characteristics of systems/equipment been determined and documented?
4.3.4	Is the development/implementation of SAMP (e.g. determination of limits, operator training, etc.) based on 'as built' characteristics?
4.3.5	Have feasible design changes of equipment been considered / implemented?
4.3.6	Have needs for special equipment to mitigate severe accidents (e.g. venting, H_2 recombiners, corium catcher, etc.) been evaluated?

State-of-the-art requirements and practices

Severe accident may create conditions that might be adverse for the usability of equipment and systems. Additionally, during the implementation of some of the strategies for returning the core or containment to a controlled, stable state, a side effect of these actions may be the creation of undesirable environments for equipment and systems. Maintaining equipment availability refers to both equipment presently in-service and equipment, which is not being used. During a severe accident, the various pieces of equipment and/or systems, which are not presently in use or required, may be useful at a later time during a severe accident. To maximise the flexibility in use of plant systems and equipment, several conditions should be considered:

- Equipment must be able to survive the environmental conditions,
- Survivability of the equipment must be not be challenged by water,
- Power supply for the equipment must be maintained, and
- Capability to repair and maintain equipment should be ensured.

Submergence in water is a special severe accident concern that must be considered with respect to equipment survivability. In severe accident management, the possibility exists for containment water levels well above the design basis or flooding the lower levels of the auxiliary building as a result of the accident or as a result of severe accident management strategies. Another important factor in equipment availability is the power supply, particularly control power under degraded DC power conditions.

The capability to repair and maintain equipment following the onset of a severe accident is important from several aspects. When arriving at a severe accident condition, it is quite likely that some of the plant equipment is not operable. During a severe accident, the potential exists for malfunctions in equipment, which is being used during the recovery. Since equipment may be used in non-standard ways for severe accident response, local access to areas may be required for alignment of valves and/or equipment maintenance.

The habitability of certain plant areas may be compromised either due to the progression of severe accident or actions taken to recover from the severe accident conditions. This may result in conditions (particularly radiation levels) in which some equipment cannot be aligned, maintained or repaired. Severe accident management decisions should take into account the habitability of plant areas in which alignment, maintenance or repair of equipment would allow for the recovery capabilities.

Information needs to be gathered on the requirements for the equipment to perform as necessary under accident conditions. These requirements are established by analysing the plant response to beyond design basis accident and severe accidents. Comparison of the expected environmental conditions and the equipment capabilities provides basis for the assessment of the equipment's capabilities to perform as required for success of individual strategies. Where possible, estimates of the operability margin of equipment beyond its design basis should be made and factored into the evaluation.

It is important that the assessment of equipment survivability is conducted based on 'as-built' characteristics of systems/equipment. This aspect is also relevant in relation to training of the staff involved in SAM. If available equipment is not able to prevent severe accident from occurring or to mitigate their consequences (either not capable of doing the required job, or not 'fit' to survive accident environment), additional equipment needs to be specified that would perform as required.

Conclusions from the assessment of equipment survivability may also lead to design changes that involve replacement of certain equipment with new one or to provide special equipment to mitigate severe accidents. There is no single approach to addressing the need for new equipment for AM. In general, whilst the implementation of an AMP may generate the requirements for limited upgrade, the requirements for major equipment changes will not be generated here. PSA (Level 1) offers a mean to decide on the need for equipment upgrades.

Current plant status

Systematic review of plant capabilities with regard to SAM has been performed. The Czech experts have not identified any need for additional hardware in relation to the implementation of SAMGs [Sy̌kora 03 a].

Capabilities of equipment considered in SAMGs have been evaluated. With regard to the survivability of equipment during severe accidents, the Czech side reported no specific problems. It should be noted that the safety related equipment currently installed at Temelín plant is qualified for harsh environmental conditions (pressures, temperatures and radiation environment consistent with the accident analysis for double ended guillotine break of the main RCS piping). These conditions are comparable to those resulting from accident involving extensive core melt in-vessel. The available SA analyses seem to confirm that.

Several minor plant design modifications have been considered in relation to SAM (as briefly described in Section 3.1.2). Preparations for implementation of these modifications are underway [Sy̌kora 03 b].

Evaluation

Based on the available information the equipment survivability aspect appears to have been addressed properly and in accordance with the current state-of-the-art. However, detailed evaluation of this aspect was not possible. This issue should be addressed within a specialized independent review (e.g. IAEA RAMP mission).

Future actions by the operator/regulator, which are recommended on the technical level to be monitored jointly in the framework of the pertinent bilateral Agreement between Austria and the Czech Republic:

- The Specialist's Team would recommend the Austrian Governement the consideration to verify that an independent external review (e.g. IAEA RAMP mission) is performed for the area of SAMG development and implementation. Such review would also assess all aspects covered in Sections 3.2 – 3.6.

3.3.4 Means for Monitoring of Plant Status under Severe Accident Conditions

VLI No.	VLI title / description
4.4.1	Has systematic analysis regarding the performance of instrumentation beyond the operational range been performed?
4.4.2	Has systematic analysis regarding the successful diversity of monitored plant physical variables been performed (e.g. identification of all available means for RCS pressure monitoring - RCS wide range, SI pump discharge pressure, RHR discharge pressure, etc.)?
4.4.3	Have any specific needs for extending the range of indications by instruments been identified and changes implemented?
4.4.4	Are there any recognised weaknesses regarding the protection of instruments against damage in severe accident conditions? Have any enhancement measures been implemented?
4.4.5	Is all the important information needed for transition from EOP to SAMP available in the MCR and in the on-site emergency centre (TSC)?
4.4.6	What is the functionality of instrumentation in station blackout conditions?
4.4.7	Is the separation of normal and emergency instrumentation / monitoring adequate?
4.4.8	Is the format of information, which is being provided to AM team, user friendly?
4.4.9	Was any instrumentation dedicated especially to SAM additionally installed?
4.4.10	Was any of computational aid (to compensate for insufficient instrumentation) developed?

State-of-the-art requirements and practices

Means for monitoring of plant status under accident conditions are required to perform the diagnosis of the severe accident progression and to select appropriate AM strategies. Usability of the existing instrumentation during a severe accident may be affected. Either the instrumentation may not survive under environmental conditions during a severe accident or the design operational range of measurements may appear to be inadequate. Severe accident may create conditions (temperature, pressure, radiation level, humidity, etc.), which might be adverse for the usability of instrumentation. Additionally, undesirable environments for instrumentation may develop during the implementation of some of the AM strategies as a side effect of these actions. Several factors determine the usability of instrumentation to monitor and forecast the progression of a severe accident, including:

- Environmental conditions,
- Presence of water, and
- Availability of AC and DC electrical power.

While instrumentation may survive under environmental conditions well beyond their design basis, it should be acknowledged that their survivability might become questionable under some severe accident conditions.

The evaluation of plant specific status relating to these aspects should concentrate on demonstrating the potential availability of and the limitations associated with all equipment and instrumentation that can be involved in the implementation of the strategies covered by the SAMG package. Relevant aspects include the identification of all alternative equipment and instrumentation, likelihood that they are available during a severe accident, clear identification of associated limitations in the guidelines, and guidance on prioritization of their usage. Important aspect is the assessment of adequacy of instrumentation ranges.

Systematic analysis regarding the performance of instrumentation beyond the operational range should be performed based on the severe accident analysis. The evaluation should include both the adverse environmental conditions and availability of AC and DC electrical power. This process may lead to the identification of potential strengths and/or weaknesses of the plant. Requirements for extending the range of measurements or instrumentation survivability may be identified and some enhancements implemented.

Current plant status

Systematic evaluation of instrumentation needs for of the implementation of SAM strategies has been performed. Example results of such investigation for the selected strategy (SAG-4 – Inject into the containment) were presented during the Prague workshop [Sŷkora 03 b].

No needs for additional instrumentation or upgrade of the existing instrumentation have been identified in relation to SAMG implementation [Sŷkora 03 a]. The existing ‘as built’ instrumentation has been found to be sufficient to support AM decision-making process (in relation to diagnostics, availability of equipment, verification of the strategy and controlled plant state).

No problems have been reported with survivability of instrumentation during SA conditions. The situation is expected to be comparable to other plants in which there is a considerable diversity of instrumentation and the majority of measurements sensors are located beyond the areas affected by the severe accidents.

Several computational aids (in the form of graphs or tables) have been developed to provide additional means to monitor those plant parameters that cannot be directly measured [Dessars 03].

Evaluation

Based on the available information, the aspects discussed above appear to have been addressed properly and in accordance with the current state-of-the-art. However, detailed evaluation of this aspect was not possible. This issue should be addressed within a specialized independent review (e.g. IAEA RAMP mission).

3.3.5 Means to Enhance Reliable Initiation of Equipment Operation

VLI No.	VLI title / description
4.5.1	Have automatic systems available for limitation of core damage and radioactivity release been implemented?
4.5.2	Are the procedures for initiation of these systems available and adequate?
4.5.3	What provisions are made for improving reliability of instrumentation? Is the concept of automatic self-testing implemented where possible?
4.5.4	Have the time margins for startup of equipment been properly determined?

State-of-the-art requirements and practices

Reliable initiation of equipment needed for the implementation of AM during an accident is an important aspect that should be addressed in the development of SAMGs. Some automatic features (e.g. automatic self-testing), which may be incorporated in the systems used to limit the core damage and radioactivity release, are of interest. Provisions should also be made for improving the reliability of instrumentation that is used in putting the equipment into operation and in monitoring its performance.

The availability and adequacy of the procedures for initiation of relevant systems credited in AMP is of high importance. Strategies credited in AMP should be based on the realistic estimation of the required time margin for startup of the equipment. This margin should be properly determined and correctly considered in the implementation of AMP strategies.

Current plant status

No additional automatic features have been considered at Temelín in relation to SAMG implementation. Reliability of the existing instrumentation is considered sufficient. No information was provided during the Prague workshop in relation to procedures for initiation of relevant systems.

Evaluation

Based on the available information, detailed evaluation of this aspect was not possible. This issue should be addressed within a specialized independent review (e.g. IAEA RAMP mission).

3.3.6 Plant Control Capabilities in Severe Accident Conditions

VLI No.	VLI title / description
4.6.1	Are there the provisions for initial warning of the MCR inhabitability? What personnel protective devices are available for MCR operators?
4.6.2	Has a reliable communication among remote locations been established?
4.6.3	Is there an adequately protected place with monitoring and control capabilities (reserve control room/panel) to be used in severe accident conditions? Are there any SA challenges identified that can affect the habitability of the reserve control room?
4.6.4	Is there a reliable power supply to important instrumentation for station blackout conditions?

State-of-the-art requirements and practices

The availability of plant control capabilities under severe accident conditions is an important aspect of SAM [IAEA 03]. Some provisions should be in place to maintain these capabilities during a severe accident. This includes the provisions for initial warning in case of the MCR inhabitability as well as the availability of personal protective devices for MCR operators. An adequately protected place with monitoring and control capabilities (emergency control room) to be used in severe accident conditions should also be available. Any SA challenges that can affect the habitability of the reserve control room should be identified and resolved.

A reliable communication among remote plant locations should be established (including the MCR or ECR, TSC, OSC, plant locations where relevant equipment needs to be restored / maintained during an accident). Diverse communication means should be available also for station blackout conditions.

Current plant status

An emergency control room (ECR) to be used in the case of MCR inhabitability is available at the plant [Sy̌kora 03 a]. The plant's experts reported no problems with habitability of ECR under severe accident condition. However, it is opinion of the PN 7 project team that the environmental conditions in the reactor building after basemat failure will preclude human occupation of the building, including the MCR and ECR. Monitoring of conditions will still be possible from the on-site underground emergency centre.

Technical means to communicate between TSC and MCR/ECR are available (verbal and visual communication lines) [Sy̌kora 03 a].

Evaluation

Based on the available information plant control capabilities in SA conditions have been addressed in relation to SAM and are in accordance with the current state-of-the-art and worldwide practice, consistent with the plant configuration. However, detailed evaluation of this aspect was not possible. This issue should be addressed within a specialized independent review (e.g. IAEA RAMP mission).

3.4 Administrative Arrangements for Personnel Response

Enhancement of the response of personnel involved in AM is one of the relevant aspects that should be considered in the development of SAM programme and SAMGs. The following subsections provide background information and discussion on the related administrative arrangements that should be in place at the plant.

Sections 3.4.1 and 3.4.2 address aspects related to the procedural framework. The current status of plant specific Emergency Operating Procedures (EOPs), which are used in response to design basis accidents, is discussed in Section 3.4.1. Guidelines, which are used in response to severe accidents (SAMGs), are addressed in Section 3.4.2. Arrangements, which are not directly related to the preparation of procedures, but which would be required to enhance the usability of the guidance on site, are addressed in Section 3.4.3.

3.4.1 Status and Features of Symptom Based Emergency Operating Procedures (EOPs)

VLI No.	VLI title / description
5.1.1	Have the scenarios contributing significantly to risk been identified?
5.1.2	What plant states are covered by the existing emergency operating procedures?
5.1.3	Have all the EOP-related symptoms properly identified? What parameters are used?
5.1.4	Have recovery actions for DBA been specified and verified?
5.1.5	Is information needed to detect level and trend of severity available to the operators?
5.1.6	Has the performance of equipment under accident conditions been verified?
5.1.7	Have the conditions for operator involvement been clearly defined?
5.1.8	Have the exit conditions and further steps been defined?
5.1.9	What was the extent of the EOPs validation?

State-of-the-art requirements and practices

The emergency operating procedures (EOPs) provide guidance to the MCR operators in response to accidents covered within the design basis (DBA). The implementation of EOPs at the plant is a prerequisite for the development of SAMGs. EOP exit conditions provide entry point to SAMGs.

The plant specific EOPs should address all potential accident scenarios that contribute significantly to the risk. They should be fully symptom-based using the symptoms that are easily verified by the existing measurements. Information should also be available to the operators to draw conclusions on the progression of accident, effectiveness of the recovery actions/ measures, as well as the severity of the current plant status. Conditions of operator involvement should be clearly defined in the procedures including the exit conditions.

The plant specific EOPs should be properly verified and validated. Effectiveness of the recovery actions specified in the procedures should be demonstrated by analysis and engineering evaluation. The performance of the equipment under accident conditions should be assessed based on best-estimate approach. Validation should be performed by practical exercises using full scope plant simulator. Training of the operators on the use of EOPs and feedback from the training process are also important aspects.

Current plant status

The process for implementation of symptom-based plant-specific EOPs was clearly presented during the Prague workshop [Hončarenko 03]. Such procedures were successfully implemented at Temelín in 1998 based on the well-known Westinghouse concept (Emergency Response Guidelines [WEC 83]).

As reported during the Prague workshop [Hončarenko 03], due consideration was given at the plant to all the above-mentioned aspects of EOP development and implementation.

The EOPs were subject to comprehensive verification (detailed walk-through method and analytical insights) and validation using CR mockup and full scope simulator. The final validation of EOPs was completed in 1999 after the full scope simulator was made available at the plant [Sўkora 01 a]. Over 20 accident scenarios were covered within the validation exercises. All findings from the V&V process were carefully evaluated, documented and resolved.

Comprehensive training was implemented at the plant for the use of EOPs. This training covered both the MCR personnel and other plant staff involved in the emergency response including plant management staff.

Feedback from operational events and from simulator training is systematically made into procedures and training programmes. All findings are documented (in the dedicated EOPs database), evaluated and resolved.

EOPs are periodically updated based on the operational experience feedback. New EOPs revision will be issued in connection to SAMG implementation [Hončarenko 03]. These procedures will define the transitions from EOPs to SAMGs. They will be finalized following the approval of the recent version of E-plant, which is currently under review by SUJB.

Evaluation

The plant specific EOPs have been implemented at Temelín. The development and implementation process was conducted in accordance to the current international practice. These procedures reflect the current state-of-the-art in this area. They provide appropriate framework for the transition to and the implementation of SAMGs.

3.4.2 Status of Severe Accident Management Guidelines (SAMGs)

VLI No.	VLI title / description
5.2.1	Have the high level AM strategies been converted into easily usable procedures/-guidelines (SAMG)? See Sections 6 and 7 for SAMGs-related quality attributes.
5.2.2	What is the actual status of SAMG implementation at Temelín NPP?

State-of-the-art requirements and practices

Under most circumstances, the preventive accident management measures included in the EOPs and implemented by the operating staff will result in plant recovery without core damage. The SAMGs are applied in the case these measures are unsuccessful and core damage occurs. Subsequent recovery actions that are covered in SAMGs (mitigative measures) place priority on containing and minimising fission product releases. The SAMGs together with EOPs provide a comprehensive procedural framework for response to accidents.

Current plant status

The work on development and implementation of plant specific SAMGs is underway [Sỷkora 03 a]. The plant specific SAMG package has already been developed (English version) and is being translated into Czech language. Verification and independent review of SAMGs is underway. The new revision of EOPs will be finalized following the approval of the new version of E-plan by SUJB. Design documentation of the related plant hardware changes (which are minor) is being prepared. Preparation of the SAMG validation process is ongoing. Plans are prepared for SAMG related training as well as for the implementation of plant hardware changes.

More detailed background information and discussion of SAMGs-related quality attributes is provided in Sections 3.5 and 3.6. This section addresses only the current status of plant arrangements relating to the development and implementation of SAMGs at Temelín plant.

Evaluation

The process of SAMG development is close to completion. Work on the implementation of SAMGs is well advanced. The SAMG development process includes all relevant elements defined in the current state-of-the-art.

It can be concluded that a due attention is given by the plant to this task albeit belatedly (IAEA Safety Series No. 110 [IAEA 93], which was issued in July 1993, called for implementation of severe accident management before commencement of Temelín operation). INSAG-10 clarified three years later that the existence of several of the elements of defence in depth does not justify operation in the absence of one element - all elements are required when an NPP is at power [INSAG-10].

Thus, strictly speaking, SAMGs should have been in place before Unit 1 was started up. Some severe accident guidance was available at the time of startup for use by the accident engineer in the TSC, but clearly this was not full accident management along the lines of the SAMGs.

3.4.3 Availability and Completeness of Training Programmes for Personnel Involved in SAM

VLI No.	VLI title / description
5.3.1	Have the training needs for different personnel involved in SAM been systematically evaluated and documented? What personnel are covered by this training?
5.3.2	Have the training programmes and schedules for training, re-training and testing of staff involved in AM been developed / documented?
5.3.3	Is the training in relation to the plant specific SAMGs addressed in the plant training programme?

State-of-the-art requirements and practices

The overall development of a plant specific severe accident management program should include tasks which are not directly related to the preparation of procedures, but which would be required to enhance the usability of the guidance on site. These include the development of a program for training the utility emergency response staff and management in the usage of the SAMG. Validation of the plant specific SAMGs against the existing E-Plan should be part of this training.

Exercise programmes should be conducted to ensure that all SAMG specified functions required to be performed for emergency response and all organizational interfaces for NPP are tested at suitable intervals. These programmes should include the participation in some exercises of as many as possible of the organizations concerned. The plan should evaluate systematically the exercises and also the regulatory body should evaluate some of them. The programme should be subject to review and updating in the light of experience gained. The attributes taken into account in the evaluation of validation exercises are further discussed in Section 3.6.3. Training of SAMG evaluation team is further discussed in Section 3.6.6.

Current plant status

Preparation of training programmes in relation to Temelín SAMG is in the planning stage [Sy̌kora 03 b]. No information was provided during the Prague workshop on the related training programmes.

Evaluation

Based on the limited available information, evaluation of this aspect was not possible. This aspects should be addressed in due time by a specialized independent review team (e.g. IAEA RAMP mission).

3.5 SAMGs Content and Structure

3.5.1 Overall Concept of SAMGs

VLI No.	VLI title / description
6.1.1	Is the symptom based concept followed in the SAMG package?
6.1.2	Are the SAMG based on a well-established generic approach? Was the applicability of this approach to Temelín evaluated based on systematic process?
6.1.3	Have a severe management closure process been established (e.g. based on NUMARC 91-04) incorporating the following elements: Utilization of the plant IPE/PSA to identify insights and enhancement that are potentially addressable by the plant specific AMP; Evaluation of the current AM capabilities and selection of safety enhancements, as appropriate; Schedule of the safety enhancements to the AMP; Establishing an evaluation process flexible for addressing new information for assessment and identification of possible plant enhancements.

State-of-the-art requirements and practices

SAMGs should convert the high level SAM strategies into easily usable procedures or guidelines. SAMGs should be fully symptom based. In general, SAMG do not attempt to diagnose the specific sequence underway but rather provide a symptom-based, structured way to determine which actions are needed in order to restore critical safety functions to prevent challenging the barriers to FP release, and finally allow the achievement of a controlled stable plant state. There should be a clear definition of the division between preventive actions covered by EOPs and mitigative actions addressed in SAMGs. Consistency with the emergency response plan should be ensured.

The procedures and guidelines must be usable and workable (the main purpose of validation being to check these aspects), and they must be presented in a user-friendly and consistent format which emergency staff can become fully familiar with and comfortable in using.

In the current practice, the SAMGs are implemented based on a well-established systematic process (e.g. [NEI 94]). This process should incorporate the following elements:

- Utilization of the plant IPE (PSA) to develop and identify a set of insights and enhancements that are potentially addressable by the plant specific AMP;
- Evaluation of the current accident management capabilities and selection of a set of safety enhancements, as appropriate;
- Schedule of the selected safety enhancements to the AMP;
- Establishing an evaluation process that is flexible for addressing new information for assessment and identification of possible plant enhancements.

Current plant status

The SAMG have been developed based on a well-established generic approach (WOG approach). The applicability of this approach to Temelín has been evaluated based on systematic process, which is consistent with the current state-of-the-art. This process includes all the elements mentioned above.

Evaluation

The overall concept of SAMGs at Temelín plant is consistent with the current state-of-the-art in the area.

3.5.2 Contents of SAMG Package and Availability of Supporting Information

VLI No.	VLI title / description
6.2.1	Has guidance been provided to all involved parts of the emergency organization (MCR, TSC, operation support, etc.)?
6.2.2	Have all the relevant phases of SAM (including the diagnosis of plant status and transfer of responsibilities) been addressed?
6.2.3	Is guidance provided for identification and optimization of strategies, actions and plant features that are to be used in SA conditions?
6.2.4	Does the provided guidance include the assessment of availability of equipment and instrumentation?
6.2.5	Are the criteria for initiating and exit SAMGs and means for checking the success of each action clearly defined?
6.2.6	What are the provisions for obtaining background, plant-specific information to support selection and implementation of SAM strategies? To what extent this information can be provided from computerized data information systems (internal network, etc.)?

State-of-the-art requirements and practices

Guidance should be provided for all involved parties (MCR operators, Technical Support Centre staff, and safety engineers). All the relevant phases of SAM (including the diagnosis of plant status and transfer of responsibilities) should be addressed.

Criteria for activation of the SAM team (TSC) as well as entry/exit conditions for each part of the guidelines should be clearly defined. Relevant decision points and the facilities to aid decision need to be identified. Plant specific set-points used in SAMGs should be plant specific and reflect the current plant status.

The actions required and equipment available to initiate actions should be defined accompanied with clear, comprehensive and plant-specific background information. Guidance should be provided on the selection of optimal SAM strategy as well as guidance on the assessment of availability and capability of plant systems to follow the different strategies. Requirements and means to override or block automatic protection signals or interlocks are in most cases not applicable to SAMGs since these actions are performed in earlier phase of the accident and covered by EOPs.

Means by which the success of each action can be judged should be described clearly. Consideration should be given to the increased possibility of erroneous readings from the instrumentation. Typically, the user is asked to confirm indication from the preferred instrumentation by double-checking against indications from other sources and by using information of different type (e.g. information on trends).

It should be noted that in addition to computational aids (CA), which can be used either manually or by means of computer, capabilities should be provided to obtain relevant support information by plant computer network. The access to relevant pre-calculated data and all related documents/reports that may be useful in selecting the optimal SAM strategy by TSC staff through computer plant network, is considered a good practice.

Current plant status

The contents of Temelín SAMG package had been well described during the Prague workshop [Sŷkora 03 b]. The contents had been found consistent with the practice presented above.

Guidance is provided in SAMGs to MCR and TSC as well as other involved parts of the emergency organization. All the relevant phases of SAM are addressed. The criteria for transition from EOPs to SAMGs are clearly defined.

Guidance is provided for diagnosis of plant status, identification and optimization of strategies, selection of potential recovery actions and plant features that are to be used in SA conditions. This guidance includes the assessment of availability of equipment and instrumentation, determination whether the strategy could be implemented using available plant features, and identification of negative impacts of the selected strategy including long-term concerns. Criteria for exit SAMGs and means for checking the success of each action are defined.

Provisions for obtaining background, plant-specific information to support selection and implementation of SAM strategies are in place. TSC personnel have access to computerized data information systems (plant data and Emergency Facility information system). Various video cameras installed in MCR, ECR and at the plant provide additional information. There is also access to an accident analyses viewer. Large variety of design documentation and all relevant procedures (both hard copies and electronic files) are available in TSC. Diversive means of communication between TSC and MCR/ECR (verbal and visual) are in place.

Evaluation

All aspects relating to the contents of SAMG package and availability of SAM supporting information during an accident were assessed based on Czech presentations during the Prague Workshop. The situation at Temelín is considered satisfactory and reflects the current state-of-the-art in this area.

3.5.3 Structure, Clarity, and Format of the SAMG Package

VLI No.	VLI title / description
6.3.1	What are the structure and the basic elements of SAMG package?
6.3.2	Are all the relevant plant areas and/or release paths (e.g. hermetic zone, non-hermetic rooms, steam generators) addressed in the strategies, where applicable?
6.3.3	What is the basis for setpoints used in SAMGs? Have the setpoints been established based on ‘as built’ plant specific features and justified by plant-specific calculations?
6.3.4	What computational aids are provided to enhance the plant status monitoring capabilities?
6.3.5	Is the clarity of presentation and user friendliness satisfactory? Are good practice and human factor principles followed?

State-of-the-art requirements and practices

Plant specific structure and format of SAMGs depend on many factors such as the definition of decision-making process and responsibilities (e.g. TSC organisation), prioritisation of SA strategies, applicability of generic strategies and studies, etc. The most known state-of-art SAMG structure is the WOG Generic structure [WOG 01]. SAMG based on this concept have already been successfully implemented in many NPPs. Brief information on the structure and contents of SAMG package prepared in accordance with this concept is provided below.

Severe Accident Control Room Guidelines (SACRG) include two parts: SACRG-1 covering initial response of MCR staff to severe accidents and SACRG-2 dealing with accident management after the TSC has been made functional.

Guidelines intended for the use in the TSC is composed of Diagnostic Flow Chart (DFC), Severe Challenge Status Tree (SCST), Severe Accident Guidelines (SAG), Severe Challenge Guidelines (SCG), Severe Accident Exit Guidelines (SAEG), and Computational Aids (CA).

Basic rules for using the SACRG are similar to those applied in EOPs. An important exception is that verbatim compliance is not required. Once SACRG-1 has been entered the actions conducted within the EOPs are discontinued. However, equipment in service at the time of transition from EOPs to SAMGs should remain in service unless the SAMG applicable instructs to the contrary.

While executing SACRG-2, the MCR staffs are also responsible for implementing strategies recommended by TSC. Typically, these actions must be approved prior to implementation by the TSC head. Exceptions are some non-critical actions necessary to control equipment conditions or to protect operating equipment.

Once the TCS enters the SAMGs (authority for this decision is specified in the ER plan), the first guidelines used are the DFC. It instructs to start monitoring of several key plant parameters (in the order specified in the DFC) based on setpoint values. Also the monitoring of severe challenges is initiated using SCTS. If a setpoint is exceeded in the DFC, the TSC implements the corresponding SAG. If a setpoint is exceeded in the SCTS, the corresponding SCG is entered. In this case, the monitoring of DFC and evaluation of SAGs is terminated.

The SAGs should provide instructions to determine the availability of equipment, positive and negative aspects of the strategy, limitations on the use of this strategy, impact of not implementing the strategy, and plant response after implementing the strategy (short-term and long-term). More than one SAG may be evaluated at a time, but the implementation of SAG strategies should follow priorities specified in the DFC.

The SCGs are similar to SAGs except that they do not contain evaluation of negative impacts. The SCG is not terminated before the relevant SCST parameter has change status (as specified in the SCST setpoints).

The SAEG-1 provides instructions for TSC long-term monitoring. These guidelines are used once any strategy is implemented (in addition to DFC and SCST monitoring). SAEG-1 instructs to monitor the parameters associated with active strategies. SAEG-2 provides instructions for SAMG termination. It is implemented once a controlled stable state is reached. Concerns to be addressed when leaving structured SAMG guidance are identified using this procedure.

It should be noted that the structure and format of SAMGs should follow general principles on human factor. Preparing appropriate instructions on writing the SAMGs (procedure writers' guidelines) at the beginning of the SAMG development project is considered a good practice.

Current plant status

The structure and format of SAMGs at Temelín were described during the Prague workshop [Dessars 03, Sŷkora 03 a]. They are consistent with the recommendations provided in the generic WOG guidelines [WOG 01] (as discussed above).

Evaluation

The structure and format of SAMGs at Temelín can be considered to reflect the current state-of-the-art practice since the authors were using all available WOG support.

3.6 SAMGs Development and Implementation

The SAM strategies, which were selected, (based on the process discussed in Section 3.2) need to be implemented at the plant in procedural form (SAMGs). The overall scope of the plant specific implementation consists of several parts:

- Definition of SAMG setpoints based on plant specific parameters;
- Development of plant specific computational aids;
- Documentation of plant specific equipment capabilities, flow path alignments and support conditions (for the attachments to the SAGs and SCGs);
- Assessment of instrumentation ranges and capabilities to define the preferred and alternate methods for obtaining quantitative information for the key plant parameters in Diagnostic Flow Chart (DFC) and the Severe Challenge Status Tree (SCST) used in the TSC;
- Integration of plant specific IPE/PSA insights into the generic SAMG;
- Introducing plant specific EOP changes based on ERG revisions for transitions to the SAMG;
- Preparation of plant specific control room SACRGs; and
- Preparation of plant specific TSC guidance (DFC and SCST, SAGs, and SCGs).

The overall development of a plant specific severe accident management program should also include several other tasks which are not directly related to the preparation of procedures, but which would be required to enhance the usability of the guidance on site. These tasks are related to the emergency preparedness and include development of a program for training the utility emergency response staff and management in the usage of the SAMG, review of and co-ordination with the existing overall ERP, and validation of SAMGs against the site E-Plan.

Correctness and quality of the plant specific SAMG package needs to be ensured by following a well defined and strictly controlled process during the development of the guidelines and by provisions for proper maintenance of the related plant documents. The following subsections address relevant aspects of SAMG implementation process.

Sections 3.6.1 – 3.6.4 focus on the relevant quality aspects relating to SAMGs development. Section 3.6.1 discusses basic QA rules that should be covered in the QA programme. The role of internal and external reviews at different stages of the development is addressed in Section 3.6.2. Validation tests that should be conducted using plant specific simulator are covered in Section 3.6.3. Provisions for SAMG revision and update are addressed in Section 3.6.4. Sections 3.6.5 – 3.6.7 are devoted to staff proficiency aspects, covering the SAM-related staffing and staff qualifications (Section 3.6.5), training process (Section 3.6.6), and feedback from training (Section 3.6.7).

3.6.1 QA during SAMGs Development

VLI No.	VLI title / description
7.1.1	What administrative arrangements have been introduced at the plant to control the process of SAMG development and implementation?
7.1.2	Has the SAMG writer team been established from plant personnel or outside sources? What was the role of plant staff in the preparation and review of the SAMG package?
7.1.3	What was the expertise and depth of knowledge of the staff involved?

State-of-the-art requirements and practices

QA applied during the development and implementation of the SAMG package should be conducted in accordance to existing QA requirements related to document control. The following basic QA rules should be followed:

- Documents should be controlled in accordance to management/administrative procedures (as defined in the overall QA program in place at the plant),
- Staff responsible for analyses should be qualified for the job and their qualification should be documented,
- All documents needed for a subsequent review should be recorded,
- Only validated and accepted references (design inputs, as-built drawings, system design description, etc.) should be used,
- Chosen strategies and assumed means (systems, structures, components) should be independently reviewed,
- Complete supporting analysis/reviews (setpoints calculations, chosen flowpaths, etc.) should be archived for a reasonable period of time.

Current plant status

It is known that general requirements of QA for preparation of safety related procedures are in place at the plant. Some information on QA aspects was provided during the Prague Workshop in the context of verification the new symptom based EOPs [Hončarenko 03].

It was stated that this process followed INPO guidelines [INPO 83-004]. Most of the QA principles mentioned above were followed. These include requirements on using qualified staff in the process of development and verification, independent reviews, strictly defined criteria for the evaluation of discrepancies, documentation of verification process, etc. Given this background, it is reasonable to assume that similar QA rules will be applied in the implementation of SAMGs.

Evaluation

Based on the information on QA principles applied at Temelín for the development of EOPs and considering the considerable experience gained in this area it can be expected that QA aspects will be satisfactorily addressed also in relation to SAMGs.

3.6.2 Internal and External Reviews

VLI No.	VLI title / description
7.2.1	Have internal and external reviews been conducted at various stages of SAMG development and implementation? Have the recommendations been implemented?
7.2.2	What mechanisms and administrative arrangements have been in place to identify potential shortcomings of SAMGs?
7.2.3	What mechanisms and administrative arrangements have been in place to ensure effective feedback from these reviews?

State-of-the-art requirements and practices

Internal and external reviews and approval are important element of QA process. It is relevant that internal reviews are performed starting from early phase of SAM development.

Identified strategies should be subject to a careful technical review by operating plant staff with broad knowledge of the plant design and safety case as well as operational aspects. Staff with good knowledge of accident phenomena and SA calculations (Department of Engineering Support) should also be involved. The existing plant Safety Committees can also play an important role.

Final instructions of the SAMG package (also the translated version) should be reviewed with respect to human performance aspects focusing on clarity of procedures (step-by-step format, use of checklists and place keeping aids, etc.). Staff responsible for this part of the review should have broad knowledge of plant design and more specific knowledge of human factor issues and 'good practise' in procedure writing.

External review should be conducted at the later stage of the SAMG development, preferably when the whole SAMG package is completed. The reviewers should have experience in the development and implementation of SAMGs. Review would also be desirable with regard to the validation exercise.

Administrative arrangements should be in place at the plant to ensure identification of potential shortcomings and effective feedback from all the reviews.

Current plant status

It was stated during the Prague workshop that the SAMG package will be subject to internal review by the team independent from the team involved in the development [Sy̌kora 03 b]. No information was provided on reviews at intermediate stages of SAMG preparation.

Evaluation

Evaluation of this aspect is of limited scope due to lack of detailed information relating to the plant specific situation. However, it can be expected that experience gained by the plant staff so far (in the development of EOPs) will be satisfactorily used.

3.6.3 Organisation and Conduct of SAMGs Validation Tests

VLI No.	VLI title / description
7.3.1	Have validation exercise been conducted? If not, what arrangements are planned?
7.3.2	Was the validation exercise properly designed in order to verify the completeness and adequacy of the guidelines, and its good perception in conditions corresponding as close as possible to real severe accidents? Which accident scenarios were selected for the validation exercise?
7.3.3	Is the documentation of the exercise comprehensive (covering the preparation, conduct, results, insights, and conclusions)?
7.3.4	Have all the lessons from the exercise properly analysed and used to propose improvements of the guidelines?
7.3.5	Did the selection of accident scenarios for validation exercise allow for testing relevant parts of the SAMG package and roles of different users?

State-of-the-art requirements and practices

The validation exercise should be properly designed in order to verify the completeness and adequacy of the guidelines. This should include demonstration of its good perception in conditions corresponding as close as possible to real severe accidents. Consideration should be given to relevant aspects including the selection of accident scenarios used in the exercise, its preparation and conduct, documentation of the exercise, and feedback of lessons learned.

The selection of accident scenarios should allow for testing relevant parts of the package and roles of different users. The documentation of the exercise should be comprehensive. It should cover the preparation, conduct, results, insights, and conclusions. All lessons from the exercise should be properly analysed and used to propose improvements of the guidelines.

Current plant status

The validation process for SAMGs at Temelín is in the planning stage [Sỷkora 03 b]. No information was provided on the details related to this process.

Evaluation

The plant has considerable experience in the validation of EOPs. The plant is also expected to receive considerable support from the Westinghouse Company, which was directly involved in the preparation of SAMGs and has both appropriate expertise in SAM and knowledge of the plant. It can be expected that this process will be carried out satisfactorily. This aspect should be subject to detailed independent review (e.g. by IAEA RAMP mission), which would be conducted when the implementation process is finalized.

3.6.4 Provisions for Systematic Revision and Update of SAMGs

VLI No.	VLI title / description
7.4.1	Are appropriate arrangements/procedures in place at the plant to review the SAMG package in the future?
7.4.2	Is there a well defined and formalized programme for conduct of reviews and their execution at regular intervals?
7.4.3	Is there a formalized programme to capture new insights, changes in technology and modifications of the plant?

State-of-the-art requirements and practices

Appropriate administrative arrangements should be in place at the plant to review the SAMG package in the future. These reviews should focus on the modifications of the plant addressing eventual changes of the plant equipment and interrelated procedures. Provisions should also be made to capture new insights and changes in technology.

The reviews should be conducted according to a well-defined and formalized programme and executed at regular intervals. The existing administrative procedures on the review of plant procedures can support this process.

Current plant status

Arrangements for periodic reviews of SAMGs and their updating are planned to be established. This issue was specifically mentioned in one of the presentation provided at the Prague workshop [Sỷkora 03 b].

Evaluation

It can be expected that appropriate arrangements for periodic reviews of SAMGs will be established in due time. A five-year SAMG update cycle was discussed at the Prague workshop, with consideration of final design changes as they occur.

This aspect should be addressed within an independent review, which would be conducted when the implementation process is finalized (e.g. by IAEA RAMP mission).

3.6.5 SAMGs-Related Staffing and Qualifications of the Staff

VLI No.	VLI title / description
7.5.1	Have staffing/qualification requirements been identified and documented?
7.5.2	Are the staffing/qualification requirements complete i.e. consider all SAM-related parts of the emergency organization (MCR, TSC, operation support, etc.)?
7.5.3	Have appropriate administrative procedures relating to this subject been developed and implemented?

State-of-the-art requirements and practices

Appropriately qualified staff should be available at the plant both to participate (or control) in the development/implementation of SAMGs and to ensure effective execution of SAMGs during a severe accident.

The staff involved in the preparation and implementation of the SAMG package should have the expertise and depth of knowledge in plant design and operation as well as in the area of SA phenomenology (including methods/techniques used in SA simulation).

Typically, the team responsible for the execution of SAMG during a severe accident is an integral part of TSC staff (see Section 2.3). The SAMG evaluation group (EG), which is responsible for the evaluation of plant conditions and determination of SAM strategies using SAMGs, is composed of 3 – 4 persons. Typically, the operation support centre (OSC) is used also in the implementation of actions during a SA. No new positions are added to OSC in relation to SAM, however, all staff involved in the execution of SAMGs should receive appropriate training (see Section 3.6.6).

It is desirable that SAMG EG staff, in particular the decision maker, have qualifications and experience in control room operation (senior reactor operator). This would make possible to discuss the actions to be taken under various SAGs and SCGs, the line-ups that have to be made, and the feasibility and execution of repairs in the plant in a proper way (considering the plant capabilities and limitations) independently from the MCR.

Staffing and qualification requirements should be clearly specified and documented for all staff involved in the execution of SAMG. Respective administrative procedures and job descriptions, which reflect the SAM related responsibilities, should be in place.

Current plant status

Information on the team involved in the development and implementation of SAMGs at Temelín is limited. It was stated during the Prague workshop that this team included Westinghouse staff and plant staff [Dessars 03]. No detailed information was provided on staffing and qualifications of this team.

Staffing and qualifications of the SAMG Evaluation Group within the TSC have not been discussed in detail during the Prague workshop. Information was given that this group includes 4 persons, namely: the Shift Supervisor, Safety Engineer, Operational Support Engineer, and Intervention Control Engineer [Sỳkora 03 a].

Evaluation

The evaluation of SAM-related staffing and qualification aspects was limited due to the lack of information. However, it is well known that Westinghouse has appropriate expertise in the area of SAM and considerable experience in SAMG implementation. Participation of the plant staff ensures good knowledge of the plant. Therefore, it can be expected that this team was properly staffed.

Some concerns can be raised in relation to staffing of SAMG Evaluation Group established within the TSC. It seems that this team is understaffed (see Section 2.4).

The SAM related staffing and qualification aspects should be addressed within an independent review, which would be conducted when the implementation process is finalized (e.g. by IAEA RAMP mission).

3.6.6 SAMGs-Related Training Programme, Training Conduct, and Training Records

VLI No.	VLI title / description
7.6.1	Have the training programme and training schedules been defined?
7.6.2	Is the training programme adjusted to the functions of the staff being trained?
7.6.3	Does the training programme for CR operators adequately cover unconventional line-ups actions which may be involved in the implementation of SAMGs?
7.6.4	What type of training is performed? To what extent the training uses practical exercises? What is the use of simulators? What software tools are available for training?
7.6.5	Does the training address relevant details of SA phenomenology?
7.6.6	Have the staff members, which are involved in SAM, been made familiar with the results of severe accident analysis conducted for the plant?
7.6.7	Is the material, which was prepared to provide basis for SAMP development, used in the training?
7.6.8	Are the emergency response staff members involved in the functional tests of equipment?
7.6.9	Is the training being conducted in accordance to the training programme and schedule?
7.6.10	What SAM-related training records are maintained?

State-of-the-art requirements and practices

The staff involved in the development and implementation of SAMGs should receive appropriate training in the field of SA and SAMG. Typically, this training is conducted at external organisations that have the required experience. These staff members become the core of the SAMG development team and later they are an integral part of SAMG EG. This staff would also be involved in the training of other staff involved in SAMG execution (ERO, MCR, OSC, etc.).

Training conducted in relation to SAMGs should include a basic training and refresher training. The training should be adjusted to the functions of the staff being trained. The existence of a well-defined and dedicated training programme is relevant

The training programme should be a proper combination of classroom training and exercises. The classroom training should address all relevant details of SA phenomenology. Presentation of the results of SA analyses for the plant is considered useful way of transferring the required knowledge. The material, which was prepared to provide a basis for SAMP development, can also be used in the training.

The training should include practical exercises. The use of plant specific simulators with severe accident simulation capabilities is desirable. Not all the plants that have implemented

SAMGs have simulators that have on-line (real time) capabilities for SA simulation. In these cases the training exercises are based on pre-calculated scenarios that are later reproduced at the training session.

The training programme should adequately cover unconventional line-ups actions that may be involved in the implementation of SAMG strategies during a severe accident. Involving the emergency response staff members in the functional tests of equipment is considered at some plants a good way of enhancing the knowledge of the plant in this context. This issue is in particular applicable to the MCR operators. Since the typical training of operating staff focuses on conventional configurations that are addressed within the DBA, this issue has been found relevant at some plants (e.g. Krško NPP).

The training programme should be documented. The training should be conducted in accordance to the training programme and schedule. The training plans and programmes as well as the execution of training and training results should regularly be evaluated, assessed and approved by the plant management.

The staff training process should be systematically monitored and all relevant training records maintained at the plant. The requirements for storage, maintenance and archiving of training-related documents and records of various types should be specified in the plant administrative procedures that apply to training in general. Typically, the documentation covered by the applicable archive system includes personnel history of training/qualification, training plans, simulator test guides, lectures, and other material used for training. Typically, the responsibility for controlling this documentation and records is given to the training department.

Current plant status

No information was provided relating the SAM-related training programme and the staff covered by this training. It was stated during the Prague workshop that such programme is under preparation [Sỷkora 03 b].

Evaluation

The SAM related training of Temelín staff is in the stage of preparation and planning. Based on the information presented at the workshop it can be expected that WOG generic training material will be used.

This aspect should be addressed within an independent review, which would be conducted when the implementation process is finalized (e.g. by IAEA RAMP mission).

3.6.7 Feedback from SAMGs-Related Training

VLI No.	VLI title / description
7.7.1	Are lessons learned from training and exercises being properly feedback into the SAMGs?
7.7.2	What are the mechanisms/provisions for feedback of lessons from training into the training programme?
7.7.3	Are the responsibilities relating to feedback process clearly assigned to individuals?

State-of-the-art requirements and practices

Experience gained during SAMG simulator exercises, which are conducted within the framework of emergency response training, should be incorporated into the training programmes. Typically, the administrative procedures describing the conduct of simulator training, which should be implemented at the plant, should also be applicable for SAM related training.

These procedures should include explicit corrective action requirements regarding the deficiencies observed during training sessions. They should be addressed and resolved in subsequent training sessions. All identified weaknesses and suggested corrective actions should be documented and analysed to determine areas, which need emphasis in future training. The responsibilities relating to feedback of this type should be clearly assigned to individuals.

Current plant status

Arrangements for feedback of experience from the simulator training into the training programmes and SAMGs were not discussed during the Prague workshop. However, these aspects were addressed in relation to EOPs [Hončarenko 03] (see Section 3.4.1). It is evident that feedback from operational events and from simulator training is systematically made into the EOPs and related training programmes.

Evaluation

Considering the existing plant arrangements for feedback in relation to EOPs and experience gained in this area so far, it can be expected that these aspects will be satisfactorily addressed also in relation to SAMGs.

4 ACCIDENT SEQUENCES AND PHENOMENOLOGY

4.1 Accident Analysis Done by PN7 Team

As the information on calculations performed by Czech side for Temelín NPP [Duspiva 01, ČEZ 02, Kujal 94, Mlady 01, Sŷkora 01 a, Sŷkora 01 b, SONS 01] was very limited, and no information on verification of these calculations was available, the PN7 team decided to perform a set of calculations for a variety of severe accident scenarios in order to evaluate issues related to severe accident management. Input data used in these calculations were based on the Temelín NPP parameters (when available) and generic WWER 1000 NPP parameters (when the Temelín specific data were not available). It was clear that this approach does not provide results directly applicable to Temelín, but it does allow for identification of potential weak points of the plant and provides a reference framework for a typical WWER 1000 NPP with Temelín features introduced where possible.

Such possibilities existed e.g. in modeling the fuel, which in Temelín is of Western production instead of typical Russian WWER fuel. Also geometrical parameters for Temelín containment and reactor cavity could be taken into account as far as known to the PN7 team. However, a number of important parameters had to be assumed, e.g. the concrete composition or the geometry of passages from the reactor cavity to the containment rooms or the characteristics and locations of hydrogen recombiners.

It was agreed that the analyses would be performed using MELCOR code, since MELCOR was known to be the principal tool used by Czech side in Temelín safety analysis in case of severe accidents and the PN7 team's analyses were aimed at providing insights into Czech calculations. Some of the scenarios calculated by PN7 team were intended as comparisons to the Czech calculations. The choice of sequences was done taking into account expected Temelín vulnerabilities, judged on the basis of our knowledge of the plant specific PSA and the existing severe accident analyses both of Temelín and other plants of similar vintage.

4.1.1 Temelín Vulnerabilities to Severe Accidents

State-of-the-art requirements and practices

The design of NPPs in the 1970s and early 1980s followed the principles of defence in depth (DID) involving multiple barriers placed between the radioactive materials and the environment. The main attention in the design and operation of NPPs at that time was directed to the first three successive levels of defence of these barriers, namely:

Level 1 Conservative design, providing margins between the planned operating conditions and the failure conditions of the equipment;

Level 2 Control limiting and protection systems, including response to abnormal operation or the indication of system failure;

Level 3 Engineered safety features and accident procedures, to control accidents within the design basis.

Later on, there have been further refinements by including the consideration of external hazards, quality assurance, automation, monitoring and diagnostic tools. Lessons learned from TMI and Chernobyl accidents showed the importance of human factors, man-machine interface, long term effectiveness of containment, need for an effective regulatory body, safety culture, and emergency planning. The current defence in depth philosophy [IAEA 88, INSAG 3, Frisch 99] gives much consideration to further levels 4 and 5, namely:

Level 4 Complementary measures and accident management to control severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents;

Level 5 Off-site emergency response to mitigate radiological consequences of significant releases of radioactive materials.

It is at Level 4 that the design of NPPs of Temelín vintage (early 1980s) did not originally provide sufficient hardware and software means for coping with beyond design basis accidents (BDBA). The NPPs designed recently take BDBAs into account both in the design and operational stage and provide means to control them so that most BDBAs do not lead to core damage. The older plants do not have such technical means, for example the measures for depressurization of the RCS are not as efficient as in new designs (for EPR the capacity of PORV is 900 t/h, for older plants about 150-200 t/h) or protection against molten corium-concrete interaction does not fully prevent hazards of basemat penetration. Nevertheless, old NPPs were often found to have considerable safety margins, especially those with large dry containments such as those in Zion or Surry NPP in the US.

The US NRC recognized this and found that in NPPs with large dry containment several issues important for severe accident management can be resolved by proper implementation of SAMGs without hardware upgrading. Such issues include hydrogen hazard, Direct Containment Heating, in-vessel and ex-vessel steam explosion. In EU the situation in NPP safety regulation area is rather different and more complicated, depending on the country. For example Spanish NPPs follow the requirements of US NRC (with some exceptions for Trillo, which is a Siemens PWR), while German and French practice is based on RSK and IPSN guidelines. In the result, both German and French NPPs with large dry containment have installed or are planning to install hydrogen recombiners, although such recombiners are not required in the US or Spanish NPPs. Of the 95 PWRs in Western Europe, the only PWRs which have not installed (or for which a decision has not already been made to install) severe accident-designed PAR units are the 6 American-designed PWRs in Spain and the 3 American-designed PWRs in Sweden. In the French NPPs the backfitting including hydrogen recombiners was decided recently and will be implemented over the next few years.

There are however some issues, where backfitting is difficult and expensive in terms of financial costs and workers' exposure. This concerns first of all the MCCI after RPV failure. For new designs such as EPR it is planned to spread the molten corium over an additional area and to install a protective zirconium layer inside the basemat, protected from above with sacrificial concrete layer (to prevent direct attack of oxide corium on zirconium) and cooled from below with a special cooling system, capable of removing decay heat from the corium. In case of old designs no such design changes are reasonably practicable.

For example in Borselle NPP presently in operation in the Netherland the reactor cavity is dry, no corium spreading is envisaged, and in case of RPV failure a melt-through of the basemat is possible within about 3 days. This is also the situation in most other PWRs presently in operation in the EU. The severity of resulting radiological releases depends on the effectiveness of fission product removal by spray systems and/or their deposition on containment internal surfaces but is largely driven by the fact that the bottom of the basemat is located several meters to tens of meters under local grade elevation. By contrast, the WWER-1000/320 plants, including Temelín, have the bottom of their basemats located about 10 meters above the local grade elevation. Calculations show that these processes (sprays and deposition processes) are quite effective, assuring dramatic reductions of volatile fission product concentration in the air within the time span needed for basemat penetration·[TACIS 02].

In the evaluation of vulnerabilities of Temelín NPP to severe accident sequences one should be aware that severe accidents are events of very low frequency, and that no absolute guarantee of safety is given for such sequences in the presently operating reactors in the European Community or elsewhere. However, the measures already implemented and planned in

EU and US NPPs are judged by the regulators to reduce the hazards to manageable proportions, both by decreasing their frequency and by mitigating their consequences. In the following chapters it will be considered whether the measures implemented and planned in Temelín correspond to the EU practice (and secondarily to US practice) in this area or not.

Current plant status

Temelín NPP has WWER 1000 type 320 reactors, and the first unit of Temelín has been under construction since 1984. It has vulnerabilities typical for reactors designed in the early 1980s (except for the elevated basemat, which is unique to the WWER 1000 design in Europe), although some of the weak points have been recently removed within the upgrading programme implemented jointly by Westinghouse and Temelín NPP. In particular, new instrumentation qualified for accident conditions has been installed, Symptom Oriented Emergency Operating Procedures have been implemented and Severe Accident Management Guidelines based on WOG SAMGs are being developed. Judging by vulnerabilities of other NPPs designed in the same period, the PN7 team identified the following issues, which require closer scrutiny:

- Protection against Primary to Secondary cooling system leakages (PRISE)
- Depressurization of the Reactor Coolant System (RCS)
- Protection of reactor basemat against penetration in the effect of Molten Corium-Concrete Interaction (MCCI)
- Hydrogen release into containment and the involved hazards of hydrogen burning
- Containment overpressurization

Each of these issues has been studied in several NPPs presently in operation in EU countries and/or in the US, and the results compared with Temelín plant status. Following sections of this report discuss detailed findings on how these vulnerabilities are dealth with.

Evaluation

Temelín vulnerabilities are similar as in other PWR NPPs designed in the same period (early 1980s) with the exception of the nature of the basemat failure vulnerability owing to Temelín's elevated basemat. More detailed discussion on whether the measures taken by the plant (already implemented and planned) to deal with plant specific vulnerabilities in accident management in a similar manner to NPPs presently in operation in the EU and US, is provided in Section 5 of this report.

4.1.2 Selection of Sequences for PN7 Analysis

The severe accident scenarios calculated by the PN7 team were chosen so as to cover the main areas of interest in SA management in Temelín NPP, at least as far as it was known before the Prague workshop held in June 2003 in which Czech side presented their SAM strategies and planned hardware improvements. The scenarios were also chosen to cover the most dominant contributors to core damage frequency identified in the updated Temelín PSA [ČEZ 02].

Since the issue of hydrogen hazards seemed the most crucial for containment integrity, the scenarios potentially involving high hydrogen releases were to be studied in detail by IRR/ARCS team, which was in charge of analyses of hydrogen distribution and hazards. Three scenarios were selected for MELCOR code analyses due to the different hydrogen combustion circumstances presented as well as their contribution to core damage frequency. These scenarios included blackout, SB LOCA with common cause failure of emergency core cooling and containment spray, and a medium PRISE scenario with extended failure to depressurize the RCS to stop the loss of primary coolant to the environment.

These three scenarios were identified in the revised PSA as the three most likely severe accidents, with a total contribution of nearly 61% of the internal events CDF. Note, however, that during the Specialist Workshop it was identified that the initiating event frequency for small LOCAs is very conservative, and thus the sequence is probably not as probabilistically important as originally thought. Note further that as a result of the MELCOR calculations performed it appears that unless the BRU-A secondary steam relief valve sticks open, the PRISE sequence is unlikely to be a core damage sequence because several days would be available for the plant staff to undertake recovery actions and terminate the sequence short of core damage.)

The hydrogen generation and distribution hazards associated with these three scenarios made them of interest with respect to hydrogen combustion hazards. The SB LOCA sequence was identified for analysis due to the hydrogen generation under circumstances where combustible conditions would likely exist in the containment (based on the Three Mile Island Unit 2 accident experience) and the containment sprays would be unavailable to mitigate fission product releases. Due to these circumstances, and due to the limited time available for the analyses, this scenario was selected in advance to be analysed with a 3-dimensional computational fluid dynamics (CFD) code (GASFLOW).

The station blackout sequence was selected for analysis based on its CDF contribution as well as the expectation that it would result in one of the largest in-vessel hydrogen releases among PWR accident scenarios. Finally, the PRISE accident was selected for analysis to investigate whether sufficient hydrogen would be released at the time of vessel failure to pose a hydrogen combustion hazard in a containment which would contain very little steam and in which containment sprays would be unavailable. The PRISE accident was evaluated assuming that the BRU-A valve functioned as designed, and also in a variation in which the BRU-A valve stuck open early in the sequence.

Additional scenarios analyzed included LB LOCA with loss of ECCS, chosen as the accident with the fastest RPV failure, for which the effectiveness of corium spreading strategy was to be evaluated. Consequently, the calculations were done in several variants, including corium ejection to reactor cavity only, or corium spreading over additional areas of 25 m^2 and 100 m^2, the latter corresponding to the area available in the room adjoining reactor cavity.

Another case considered was total loss of feed water (LOFW) with EFWS not available and SB LOCA 50 mm with loss of ECCS and EFWS. These cases were studied to find out the effectiveness of various methods of RCS depressurization when EFWS is not available, and so RCS cooling through SG depressurization is of limited use. Several strategies of SAM were studied, with RCS depressurization by means of opening PORVs on the secondary side of SGs, by using the Emergency Gas Removal System from the RCS and by opening PORV on the pressurizer. The case with no SAM leading to high pressure scenario with RPV failure under high pressure was also studied.

In total the calculations conducted in PN7 teams were meant to provide insights relating to the following aspects:

- Effectiveness of corium spreading strategy
- Effectiveness of various methods of RCS depressurization
- Plant response to hydrogen releases to the containment
- Containment overpressurization hazards in case of low pressure and high pressure RPV failure scenarios.

4.1.3 Overview of analyses

The cases analyzed cover a wide spectrum of severe accident conditions and can be therefore of interest in reviewing the results obtained for Temelín NPP. The main findings of these analyses are presented in the Annex, in Table A.1. The first set of calculations presented in Annex A (Table A.1), was performed by Czech TSO. It includes analyses of PRISE with two variants (with and without thermal creep modelled), analyses of LB LOCA with loss of ECCS also performed in two variants (with and without hydrogen detonation), the analysis of blackout, and finally the analysis of LB LOCA with late recovery of ECCS [Kujal 03, Pazdera 03].

One of the results reached in Czech calculations is that the principal hazard to containment integrity is the possibility of basemat penetration by molten corium. The strategy of corium spreading and flooding of reactor cavity with water before the RPV failure is evaluated in the Czech calculations to result in significant slowing down of the MCCI processes and the Czech authors claim that eventually the concrete penetration can be completely stopped [Sy̌kora 01b, Sy̌kora 03].

However, in the light of present day knowledge this claim cannnot be proved. The large-scale experimental work aimed at clarification of MCCI processes is still going on. The existing experimental evidence (from the MACE smaller-scale program and the WETCOR program) does not support the thesis that concrete penetration can be stopped since melt quenching was not achieved in any of these experiments [SKI 2000]. MELCOR code calculations carried out by the PN7 team also suggest that concrete penetration will finally occur.

The set of calculations performed within TACIS programme covered all important severe accident sequences and was used to develop PSA level 2 for Balakovo NPP [Morozov 03]. The results of this work show robustness of large dry containment typical for WWER 1000 units and resistance to hazards of DCH or hydrogen burns, and high effectiveness of WOG SAMGs, which reduce very much the frequency of unmitigated severe accidents.

The analyses performed by PN7 teams are discussed in more detail below in Section 4.2, but generally their results are in good agreement with the Czech and TACIS conclusions. Some of the PN7 calculations were performed to parallel Czech calculations, which were presented during meetings held before the Road Map process was initiated. These calculations were done with limited nodalization of the containment (11 or 14 nodes in most cases, 54 nodes in one case; compared with 6 nodes most of the Czech calculations). Not surprisingly, similar results were noted owing in large part to the similar modeling concepts applied. The PN7 analyses broadly show good resistance of large dry containment to severe accident hazards with the exception of basemat penetration, which remains the weak point of Temelín NPP, and with the possible exception of the potential (for limited periods of time) for energetic hydrogen combustion modes.

Additionally to MELCOR analyses, based on modelling the containment with only a limited number of volumes, a 3-D analysis with GASFLOW has been performed to show a different picture of hydrogen distribution in the Temelín containment, here subdivided into a fine mesh of 52,080 volumes. The results of this analysis are discussed below in Section 4.3.2 and 5.5.

4.2 Discussion of selected accident sequences

4.2.1 PRISE Sequences

PRImary to SEcondary leakages (PRISE) are those occurring due to Steam Generator tube ruptures (SGTR) or in WWER units due to primary collector leakages to the secondary side of SG. They are potentially dangerous events, because due to leakages in steam generators two barriers preventing fission products releases – namely RCS boundary and reactor containment – are lost at the start of the accident, and the two other barriers – fuel pellet matrix and fuel cladding – can be lost in the course of the accident, unless it is effectively mitigated. Moreover, in this accident fission products are released together with steam, which results in their long range carry-over, with maximum deposition densities in the distances of 30÷50 km [SONS 01]. In order to gain the necessary insights concerning strategies to be followed after PRISE a set of calculations was undertaken within PN7.

Two variants of the PRISE sequence were calculated. In the first case, the scenario was modeled assuming that the BRU-A valve performs without failure. The Czech experts claim BRU-A qualification for two-phase and water flow, although questions have been raised in this regard in an earlier Roadmap project [UBA 2003]. As a result, a second calculation was done assuming that the BRU-A valve fails open after it passes water flow. The results of these two calculations are markedly different, thus emphasising the importance of BRU-A qualification for water and two-phase flow.

It is worth noting that even if the valve is assumed to be qualified, the valve is demanded to be opened and closed many dozens of times during the course of the transient. Even if fully qualified, the valve has a demand failure rate for opening and closing which, if applied dozens of times in the course of a transient results in a non-negligible conditional probability that the valve will fail. A typical failure-to-close rate is about 5×10^{-3} per demand, so if the valve is demanded a few dozen times, the failure rate for the entire scenario is quickly above 10%. Thus, even if - as asserted by the Czech experts - the BRU-A is fully environmentally qualified, due to the large number of opening and closing demands during the PRISE accident, there is a significant chance that the valve will fail open anyway.

The base case calculation, with the BRU-A assumed to perform correctly, results in an extremely long accident progression. Assuming no actions to either depressurise the reactor coolant system or provide additional water to the sump to assure extended high pressure injection (there are three 400 m^3 tanks of borated water which can be pumped at 50 m^3/h to the containment sump to provide for additional HPI water supply) [Czech 01], the MELCOR 1.8.5 calculations indicate that approximately five days are available for recovery actions before the core starts to heatup leading to severe accident conditions.

During such an extended period of time (five days, compared with a number of other severe accident sequences which go to core damage between one and twenty hours), it is nearly inconceivable that there would be a failure to depressurise the reactor coolant system and provide makeup water to the containment sump. Even if the plant staff develops a fixed mindset about what is going on – and bear in mind that the symptom-oriented EOPs and SAMGs are designed to avoid such a condition – with five days available it is hard to imagine that assistance would not be forthcoming from the regulatory authority, from the Czech research center (NRI Rez), and from international entities (e.g., Westinghouse, EU research and regulatory organisations, the OECD Nuclear Energy Agency partners of the Czech Republic, and the International Atomic Energy Agency partners of the Czech Republic) that would allow the overcoming of this mindset and the termination of the accident scenario well short of core damage within five days.

Thus, it is concluded that the PRISE accident with nominal performance of the BRU-A valve on the main steam line of the affected steam generator would be very unlikely to result in core damage.

When the BRU-A is assumed to stick open when first challenged with a water discharge, the calculation resulted in a severe accident with reactor vessel failure occurring between 18 and 19 hours after the start of the accident assuming failure of the operating staff to implement EOP provisions to depressurize the primary coolant system. This outcome shows the sensitivity of the progression of the scenario to the status of the BRU-A valve.

The updated Temelín PSA estimated the frequency of the base case scenario at

$3{,}09\times10^{-6}$ [1/a] [Mlady 03 a]. However, as noted above, it is considered very unlikely for the base case scenario to result in core damage due to the extremely long period of time (over five days) before core heatup leading to severe accident conditions occurs.

Thus, the PRISE sequence leading to core damage would have frequency of the updated Temelín PSA value times the conditional probability of BRU-A failure (i.e. $3{,}09\times10^{-6}$ [1/a] times 0,1 for a frequency of $3{,}09\times10^{-7}$ [1/a]). Even for this frequency, it is not clear that the full measure of SAMG actions has been considered in the updated PSA. Several SAMG strategies appear to provide means for either terminating the PRISE accident progression outright (SAG-2, "Depressurize the RCS"), extending the time available for recovery (SAG-3, "Inject Into the RCS", or SAG-4, "Inject Into the Containment"), or at the least mitigating the accident consequences (SAG-1, "Inject Into Steam Generators").

It is recommended that the scenario quantification in the PSA be revisited by ČEZ to consider: (a) the thermal-hydraulic accident progression calculations for the sequence, (b) the conditional probability of BRU-A failure to close given numerous actuations (as indicated by the thermal-hydraulic calculations), and (c) the influence on scenario frequency resulting from implementation of the SAMGs.

4.2.2 Station Blackout

This scenario is due to loss of AC power from internal and external sources, and failure to start at least one of the three Diesel Generators. This loss of power leads to the failure of Emergency Feedwater System (EFW), High Pressure and Low Pressure Injection System (HPIS, LPIS), Containment Spray system (CSS). Passive safety features are available (Safety Injection Tanks (SITs), Safety Valves (SV) on the pressurizer (PRZ) and on Steam Generators (SGs) and systems that receive power from accumulator batteries (BRU-A, the emergency gas removal system (EGR), the containment isolation valves) remain available.

The time for battery discharge determines how long the operators can monitor plant status with their instrumentation and perform some limited actions, which require only battery power (but no AC power). When the batteries are run out, the control over the plant is lost. Therefore, the battery capacity is of high importance for severe accident management. In the MELCOR 1.8.5 calculation in PN7 reported below, the design value of 1 hour was used as the time of battery depletion (information presented at the Prague Workshop indicates that with proper operator response to conserve battery power, a time of 3÷4 hours can be achieved).

During blackout in a WWER 1000 NPP the heat from the core is removed by natural circulation of the coolant to the SG and then by evaporation of water in SG and steam release through BRU-A to the environment. It is worth to be noted that failure of secondary depressurization capabilities is unlikely (should the BRU-A fail closed in a main steamline the main steam line safety valves, two per steam line, would lift at a somewhat higher pressure to release steam to the environment.)

The MELCOR calculation indicates that the first BRU-A opening occurs within 200 seconds of the loss of power. The loss of water from the secondary circuit during its depressurization through the BRU-As (which open at 7,3 MPa) and the absence of feedwater injection to the SGs leads eventually to full dry-out of the SGs at 7100 seconds (118 minutes). This sharply decreases heat removal from the RCS. Then the heat is removed by primary coolant heatup that leads to primary pressure increase above 17,6 MPa and steam dump into containment through pressurizer relief or safety valve. The pressuriser fills and begins water relief at about 8300 seconds (138 minutes) after the loss of power. As the pressurizer relief ("barbotage") tank fills up, the tank membrane fails, leading to the first significant release of water and steam into the containment.

After about 10 000 seconds (166 minutes), core uncovery begins. The start of metal/water reaction between the hot zircaloy fuel cladding and steam begins at about 14 700 seconds (4 hours). Core slump to the lower head occurs at about 19 800 seconds and RPV failure occurs soon after at 20590 seconds (5,7 hours). RPV failure occurs at high pressure due to the unavailability of means by which to depressurise the reactor coolant system below the pressure of 1÷2 MPa at which pressure melt ejection and direct containment heating can occur.

At the time of RPV failure by melt-through, the sudden reactor pressure drop causes the SITs to discharge their fluid to the reactor vessel, and part of corium is cooled with water causing it to remain within the reactor vessel. After vessel failure, any remaining water from the SITs enters the reactor cavity through lower head failure area, together with molten corium. Corium remaining in the reactor vessel is heated up again and after repeated melting at high temperature reaches the reactor cavity in the form of low pressure "pours" through the RPV failure location.

At the time of RPV failure, the pressure in the reactor cavity was estimated to rise to somewhat less than 0,8 MPa. As the containment (and internal structures) are designed for the design basis accident (large LOCA) pressure of 0,49 MPa, this pressure rise should not (considering normal conservatism in design) present a threat to the structural integrity of the reactor cavity. Also at the time of RPV failure, the average containment pressure rises sharply to about 0,45 MPa as a result of melt ejection and direct containment heating. This pressure is below the design pressure of the containment, and would not be associated with a potential for containment failure. However, without heat removal from the containment, containment pressure starts a slow rising trend until about 32 hours when the containment pressure would rise above 1 MPa, which is close to the estimated median failure pressure of the containment. At around this time or soon after, containment failure would be expected to occur due to overpressure resulting from the lack of containment heat removal.

The calculation above assumed unrecovered loss of AC power. The sequence frequency estimated in the updated Temelín PSA implies that the station blackout condition lasts long enough for core melt and RPV failure to occur (i.e., between 5 and 6 hours). Continued station blackout conditions in the time after this would in reality be at a lower frequency of occurrence. Nearly all station blackout conditions would be expected be recovered within 32 hours (indeed, the 1995 PSA indicated that 98% of offsite power losses are recovered within 10 hours).

Only very extended grid failure, owing perhaps to an external hazard or a large area grid failure, would extend a loss of offsite power to a duration approaching 32 hours. In addition, a failed diesel generator would be expected be repaired within such a long time frame. Note that each Temelín unit has three diesel generators of its own, and in addition there are two shared non-safety diesels, which can be aligned to provide power at a safety bus in either unit (only one of these five diesels needs to be recovered in order to provide power to one train of safety systems).

Thus, it is likely that power recovery would occur in the time frame between RPV failure in the 5÷6 hours time frame and containment overpressure failure in the 32-hour time frame.

The question then becomes what happens when the power is recovered. If the containment spray pumps are not disabled before power is recovered, the containment spray pumps will start on a high containment pressure signal. At times after about 3 hours, the containment is steam inerted in the station blackout scenario. Startup of the containment sprays will rapidly "de-inert" the containment atmosphere due to steam condensation, which will rapidly increase the percentage concentration of hydrogen and oxygen. The outcome of such a situation depends on how much oxygen and hydrogen remains in the containment atmosphere (allowing for accident progression and hydrogen recombination by the PARs, which depletes both hydrogen and oxygen from the containment atmosphere).

Experiments have been conducted on the outcome of such conditions as part of the review process for the Combustion Engineering System 80+ design certification process. The experimental setup included the presence of hydrogen igniters, which ensured constant availability of hydrogen ignition sources. Eight tests were conducted under well-mixed conditions and three under stratification conditions. The tests were conducted with sufficient hydrogen to represent an average concentration of 13,5% under dry conditions. In the eleven tests, no detonations were observed. All tests ended in multiple deflagrations or a single deflagration, which in no case was threatening to containment integrity [Blanchat 97]. While there are differences in geometry between the experimental setup and the WWER 1000 design, the experiments suggest that detonations are not very likely outcomes when a steam-inerted, hydrogen-containing atmosphere is subjected to containment spray. These experiments lend credence to the modeling of hydrogen combustion under steam-inerted, containment spray recovery conditions as a deflagration or series of deflagrations, depending on the conditions in the various containment compartments.

Calculations of this situation with MELCOR within PN7 (for a LOFW rather than station blackout, however) suggest that the resulting pressure increase from deflagration under rapidly deinerting conditions are below or slightly above the design pressure for the containment (that is, less than 0,49 MPa), and therefore should not pose a threat to containment integrity (the minimum containment failure pressure, corresponding to the 5^{th} percentile of the failure pressure curve, is 0,8 MPa).

4.2.3 Other Accident Sequences

Besides blackout, other accident sequences involving simultaneous beyond design basis failures in NPP operating systems and safety systems can result in loss of heat removal capability from the core. Usually considered BDBAs are SB LOCAs with simultaneous loss of ECCS and accidents initiated by loss of all feed water to steam generators (LOFW), including auxiliary feed water system and emergency feed water systems (EFWS).

According to recent French estimates, the cases with maximum hydrogen releases are those for SB LOCA (25 to 75 mm diameter break) with CSS available but with no Safety Injection, and LOFW with CSS but no Safety Injection. Within PN7 project both these types of scenarios were calculated, taking into account both in-vessel and ex-vessel phases and considering several possible SB LOCA break sizes. In all transient sequences excluding LB LOCA the question of effective depressurization of the RCS is of primary importance, just as it is for blackout sequences. For checking the plant ability to depressurize the RCS before the RPV break, the sequences of SB LOCA were chosen to have the size of the break at the lower limit of break spectrum, namely 15 mm.

4.2.3.1 Total Loss of Feed Water scenarios

The purpose of the analysis of a total loss of FW was to explore SAM action – primary depressurization to mitigate and if possible – prevent core degradation. In case of LOFW there are two possibilities of RCS depressurization – by opening PRZ PORV or by using emergency gas removal (EGR) system. Both systems are safety graded and operable from the control room. The opening of the PORV will result in a rapid increase of pressure in the relief tank, failure of the rupture disks and LOCA to the containment, which is not acceptable for an operational transient. On the other hand, the gases removed by EGR are directed to the gas purification system, so they remain contained and do not contaminate the containment atmosphere. Therefore, using the EGR system is the better solution.

AM procedure investigated was a primary depressurization with the gas evacuation (emergency gas removal) lines from the top of the RPV and from the pressurizer. The scenario was run with availability of all ECCS systems. The criterion for initiation of the SAM procedure was core exit temperature exceeding 650 °C. The analysis assumed total loss of FW (both normal and emergency) and thus – loss of heat sink. Make-up system was also assumed to be lost. All 3 HPIS trains, 4 SITs and all 3 LPIS trains were available. To establish water injection to the primary system it is necessary to achieve depressurization down to the HPIS injection pressure.

Three cases were analyzed:

- Scenario with depressurization using BRU-A as specified in EOP (0,5 h after the accident), then depressurization with gas evacuation system (RPV+PZR) and PZR PORV,
- Scenarion without opening BRU-A, depressurization with gas evacuation system (RPV+PZR) and PZR PORV ,
- Scenario with loss of all active ECCS, no depressurization.

Initial cycling of BRU-A till 30 minutes depletes significantly the SGs secondary inventory (to 1 m below the initial level). The forced opening of BRU-A between 30 minutes and 60 minutes reduces very fast the SG secondary coolant level by another 1 m in about 8÷10 minutes. Thereafter, SGs do not influence the primary pressure response due to loss of the secondary inventory and loss of primary-to-secondary heat transfer.

Shortly after starting the depressurization with BRU-A (at time 0,53 h) the primary pressure is reduced for a short period of time below the shut-off head of the HPIS pumps and there is injection for a short period of about 480 seconds.

Thereafter the primary pressure increases due to the loss of heat sink. At time 2,13 h the active core starts to uncover. Core fully uncovers in 865 s and stays uncovered for almost 1300 s. The operator starts primary depressurization by using Emergency Gas Removal system and PORV at 2,36 h. At time 2,55 h as a result of the primary depressurization by the operator the HPIS starts steady delivery of water to the primary system.

Peak cladding temperature reaches 1540 K at time 2,6 h after the accident - fuel heatup is sufficiently high to fail the cladding. Gap release starts at time 2,58 h after the accident and fuel cladding tightness is lost in all fuel elements.

In 2200 s the primary pressure goes down to 60 bar - SITs start injection of water to the primary system at 2,84 h. Injection from SITs goes for about 540 s. At around 2,51 h and 2,73 h there are observed spikes in the primary pressure due to the sharp evaporation and increase of the steam in the primary system by the heated fuel in the reactor core and at the same time - injection of water to the primary system. At time 276 h the core is covered again and the fuel is cooled down.

The total amount of oxidized Zr is 121,6 kg, representing 0,454 % of the total mass of zirconium in the corewith production of 5,4 kg of hydrogen. After depressurization of the primary system the injection of water by the HPIS brings the reactor into a stable cooled down state and the accident is successfully managed.

In the case without BRU-A opening the sequence is similar but the timings are slightly different and the amount of Zr oxidized is smaller. Peak cladding temperature reaches 1400 K at time 3,54 h after the accident - about 1 h later than for the case with opening of BRU-A. The heatup of the fuel is sufficiently high to fail the cladding - gap release starts at time 3,48 h after the accident. However, the fuel does not melt and the releases of fission products from fuel pellets are small.

H_2 generation for this case is 1,41 kg as opposed to 5,38 kg of hydrogen for the case with the opening of BRU-A. The peak of the fuel temperature is about 50 minutes earlier and the peak value is 140 K higher for the case with opening of BRU-A. The peak fuel temperature for the case with BRU-A is 1540 K, and for the case without opening of BRU-A – 1400 K.The calculations showed that in case of entering SAMGs at core exit temperature exceeding 650 °C and using EGR and PORV the accident can be successfully managed without core melt.

The third case calculated was that of LOFW with loss of all active ECCS systems, and no SAM actions. In such a case the sequence leads to core melt and RPV bottom head failure under full pressure. Total H_2 generated in-vessel is 544 kg, bottom head is penetrated at 6,8 h, and total corium mass ejected is 147 t.

For this case – LOFW without SAM actions – two sets of calculations were performed, the first for generic data on basemat concrete composition, the second for updated concrete composition, which was made known to the Austrian side after the Prague Workshop. The difference consisted mostly in the fact that the concrete used in Temelín for the basemat has no carbon content and so would not produce noncondensible gases CO and CO_2 during MCCI. Moreover, reduced PAR capacity was assumed, taken according to the latest data of the PAR producer.

The results for the updated concrete composition showed that in the case without SAM actions the upper layer of serpentinite concrete would be penetrated after 16 hours since the beginning of the accident with the average axial penetration rate of 7,6 cm/h, and the amount of hydrogen generated due to MCCI in serpentinite concrete would be about 850 kg. After that the molten corium would attack the lower base concrete layer. The rate of axial penetration would be 11,8 cm/h and the concrete basemat would be penetrated within about 42 hours after the start of the accident. The amount of hydrogen generated due to MCCI during base concrete penetration would be about 1100 kg.

The calculations covered also evaluation of containment pressure and hydrogen concentration during the accident, assuming division of containment into 11 control volumes with uniform gas concentration in each volume. The pressure peaks in the early phase after the RPV break in the reactor cavity were reaching 0,4 MPa, and in the long term the maximum pressure in the containment was kept at about 0,38 MPa, so under the design strength of the containment. Due to large steam release at the moment of RPV failure the atmosphere in the containment would be inerted. Later on the fraction of steam decreases and at about 16 hours there will be a possibility for hydrogen deflagration in the containment. In the long term the oxygen depletion in the result of hydrogen recombination in PARs will bring the steam-hydrogen-air mixture to inerted conditions, but this process is slow due to low capacity of PARs and the depletion threshold is reached after about 35 hours.

In the reactor cavity the concentration of H_2 is high, but after penetration of the RPV bottom head a large amount of water from SITs enters the cavity. The evaporation of this water increases the steam concentration up to almost 100 % and this inert condition inside the cavity continues for the first 7 hours of the MCCI. After inertization of the cavity the cavity atmosphere has insufficient O_2 for deflagration. Later on, the depletion of oxygen provides for inerted conditions. The peak recombination rate of one PAR is about 0,18 g/s for the H_2 and 0,1 g/s for the CO. The H_2 mass recombined until 31 hours of the accident in one PAR situated in the optimum position is about 14,3 kg and that of CO – 3,34 kg. Of the total amount of hydrogen produced in-vessel and ex-vessel about 315 kg H_2 (11,5 %) are recombined within the time covered by the calculation. The rest of combustible gases stays inside the containment.

The penetration of the bottom head of the RPV creates very high steam concentration also inside the containment compartments. In fact after the onset of MCCI the containment atmosphere is inert. 16÷17 hours after the accident the H_2 concentration exceeds the critical value of 10 % where self-deflagration is possible. The steam and gas distribution inside the containment is not fully uniform. There is significant steam and H_2 stratification predicted by the MELCOR code. The containment atmosphere is still inert by the high steam concentration. The condensation of the steam on the containment walls slowly decreases the steam concentration and 17 hours after the accident the steam concentration in the containment dome goes down below 55 %. The O_2 concentration is sufficient for deflagration (above 5 %). All this provides conditions for deflagration of the hydrogen. The operation of the recombiners slowly reduces the amount of O_2 in the containment. Due to the small capacity of the recombiners the reduction below the minimal 5 %, at which deflagration is still possible, is not earlier than 34,8 hours after the accident.

The steam condensation inside the ECCS compartment is faster. This compartment is not inert during all of the accident - steam concentration is below 55 vol.%. At 15 hours after the accident the H_2 concentration in the ECCS compartment reaches 10 vol.%. This creates conditions for deflagrations inside it. During the next 6 hours multiple deflagrations are observed in this compartment. The temperature peaks during the deflagrations are about 680÷690 ^{O}C. The pressure oscillations in the containment are very mild because the deflagrations do not propagate to other larger compartments due to the inert conditions in the rest of the containment. These multiple deflagrations slowly deplete the oxygen in the containment. Towards 35 hours after the accident the O_2 volumetric concentration in the dome is below 3 vol.%. The steam concentration is about 51,3 vol.% and the H_2 concentration is above 13,5 vol.%. Due to the lack of sufficient oxygen there are no more deflagrations.

Although there is no need for spraying because the containment pressure is low, a calculations run was made assuming that the spray system is started to see what would be the impact on the containment integrity if deflagration occurs at this moment.

The spray condenses rapidly the steam in the dome and in the SG boxes, so that the concentration of steam is reduced to 43 vol.%. This results in an increase of the O_2 concentration above 5 vol.%. The H_2 concentration is also increased reaching 16,3 vol.%. This creates a large deflagration inside the containment, with the pressure peak of 5,40 bar. The temperature reaches 757 ^{O}C. After this major deflagration the oxygen is depleted below 2,1 vol.%. The containment parameters during that deflagration do not exceed the lower limit of the containment strength, so there is no danger of containment early failure.

On the other hand the penetration of the basemat due to MCCI propagates and the basemat failure can be expected before 40 hours. Thus the main hazard in case of LOFW would be connected with containment basemat melt-through in a situation where the concentration of H_2 in the cavity atmosphere is high.

4.2.3.2 SB LOCA

4.2.3.2.1 Evaluation of Depressurisation Behavior in Case of 15 mm LOCA

In order to find out the possibilities of RCS depressurization by Temelín systems in case of a LOCA with smaller leaks, the consequences of SB LOCA 15 mm were calculated. In these analyses total loss of FW and failure of all 3 HPIS trains was assumed. In addition to the loss of HPIS, also the loss of high-pressure boron injection pumps was assumed. The 4 SITs and all 3 LPIS trains were assumed to be available. Thus, to establish water injection to the primary system, it was necessary to achieve depressurization down to the SITs pressure and thereafter – down to the LPIS injection pressure

AM procedure investigated was RCS depressurization using the emergency gas removal system (EGRS) lines from the top of the RPV and from the pressurizer. At late stages of the accident progression it was necessary to depressurize the primary system also by opening of the pressurizer's PORV in order to establish stable LPIS injection and cooling of the core. The criterion for initiation of the SAM procedure was core exit temperature exceeding 650 °C.

The calculations showed that the Emergency Gas Removal System allows to depressurize the RCS and prevent RPV failure. This - as ETE stated - is the case only, if the criterion for entry into SAMG is set low (650 °C) and the achievable flow rate though EGR is high (above 30 kg/s).

However, in order to reach the stable safe state it was necessary also to open PORV in the last stage of the accident. This indicates that the EGR system may be unable to depressurize RCS by itself, without using PRZ PORV. Moreover, in the calculations the flow rates of EGR system were distinctly higher than the upper limit established in rough estimates for that system, namely 30 kg/s.

Therefore, one more calculation was run with the same initial conditions, but with the maximum EGR system capacity limited to 20 kg/s and with entry into SAMG criterion established as 650 ^{o}C. In this case the operation of EGRS was not sufficient to depressurize the primary system. The accident evolved into a severe accident with core melting.

Finally, one more case was studied, namely with total loss of FW (both normal and emergency) and thus – loss of heat sink, but with all 3 HPIS trains available, all 4 SITs and all 3 LPIS trains available. As before the loss of high-pressure boron injection pumps was assumed. The entry point to SAMGs was set at core exit T > 650 ^{O}C.

The calculations showed that with entry point 650 ^{O}C, with EGR capacity limited to 20 kg/s and with HPIS available it is possible to succeed in a partial recovery and cooldown of the core. However, also in this case it will be necessary to depressurize with PORV to achieve stable cooling of the core.

4.2.3.2.2 Severe Accident Calculation for 50 mm Small LOCA

A severe accident scenario involving a small LOCA was also modeled in the PN7 project. This scenario, which is identified in the PSA as the most likely severe accident sequence (but see comments below in this regard), results from a small LOCA with common cause failure of high and low pressure injection and containment sprays. The primary and secondary pressure relief valves (PORV and BRU-A valves) were assumed to be available as was the charging pump system (consistent with the PSA definition of the scenario), so the accident progression was modeled with correct operator action to depressurise the reactor coolant system. Thus, at the time of vessel failure, the pressure in the primary coolant system was low and high pressure melt ejection from the vessel was avoided.

The small LOCA was modeled as occurring in the steam generator box where the pressurizer is located, and the break was modeled as occuring on the primary loop which includes the pressurizer (this was done in order to evaluate the effect of the break location on the "loop seal" in the affected loop).

The operators were modeled as correctly connecting the TB10 tanks to the charging pumps. This provides additional charging pump water inventory to extend injection from the charging pumps. The charging pump inventory was calculated to be exhausted by 5,2 hours. As a result of operator action to depressurize the reactor coolant system, the hydroaccumulators (SITs) also discharged their inventory to the reactor coolant system. This action was completed within 3,2 hours of the start of the sequence. At 5,9 hours, continued loss of coolant inventory from the break results in the onset of core uncovery. Metal-water reaction leading to the production of hydrogen commences at 7,2 hours. Reactor pressure vessel failure was calculated to occur at 8,3 hours. Reactor coolant system pressure remains low through the time of vessel failure as a result of actions by the operators (consistent with the EOPs and SAMGs) to depressurize the reactor coolant system.

The small LOCA sequence was also run without operator action to depressurize the reactor coolant system In this case, the core damage and vessel failure events were considerably later (with vessel failure after 13 hours). However, the primary system pressure is borderline with respect to melt ejection, and since more of the energy in the primary system is transferred through the break to the containment (instead of to the environment via the BRU-A through depressurization efforts by the operators), the containment pressure is considerably higher in this case. The containment pressure was calculated by MELCOR to be near 0,8 MPa at the time of vessel failure. As estimated by CEZ [Sýkora 01 a], at a pressure of 0,8 MPa the conditional probability of containment failure is about 5 % (one chance in 20).

4.2.3.2.3 Detailed Hydrogen Distribution Modeling With GASFLOW II

The in-vessel MELCOR 1.8.5 modeling of the small LOCA sequence (base case) was used as input to the GASFLOW II code to allow for a more detailed modeling of hydrogen combustion potential by estimating local, heterogeneous gas distribution within the containment. It is important to understand how MELCOR models hydrogen combustion, in order to appreciate how the GASFLOW code identifies conditions which are so different from those encountered with MELCOR.

MELCOR is a "lumped parameter code" – it does not model hydrogen distribution in a mechanistic manner. The MELCOR code requires the code user to define calculational nodes, which are assumed to be homogeneous. Such "nodalization" is an art, more than a science, because a calculational cell, which has homogeneous conditions in one part of an accident may not be homogeneous in another part of the same accident scenario. If the nodes are made too large, anomalous results can occur (if the nodes are made too small, calculation times become very long).

For example, take the case where a small opening into a large compartment conveys a concentrated hydrogen gas stream. In the MELCOR code, the amount of hydrogen conveyed into the large room during the time step of the code is calculated. Then that hydrogen is instantaneously and homogeneously mixed by the code into the entire volume of the compartment. Then the code checks to see if minimum combustion criteria are met. If not, no burn occurs. If so, then the code models a burn as occurring, *irrespective of whether an ignition source is present*. Moreover, the code can only model deflagrations or diffusion flames - the code is inherently incapable of modeling flame acceleration, deflagration-to-detonation transition, or global detonation.

What may happen in the real (three-dimensional) world is as the hydrogen gas enters the compartment; a region of combustible or maybe even detonable mixture is created. When homogeneously "smeared" over the entire compartment volume by the MELCOR code, non-combustible conditions can result, but in reality some fraction of the volume is combustible and perhaps even detonable.

But <u>any</u> time the MELCOR code calculates that minimum combustible criteria are obtained in a time step, a burn is initiated. The hydrogen burning is in reality a stochastic process. Hydrogen does <u>not</u> always burn when combustible conditions exist. Hydrogen does not always detonate when detonable conditions exist. But MELCOR is only capable of identifying that the conditions are met, then numerically deflagrating the hydrogen or flagging the attainment of a detonable concentration. Experiments have been conducted in which decidedly detonable conditions existed (e.g., 24 vol.% hydrogen with sufficient oxygen to support combustion and insufficient steam to prevent combustion) without a detonation occurring.

If a code user gets results from a lumped parameter code that indicate that a number of mild deflagrations occur over a period of minutes or hours and no pressure threatening containment was calculated, what is the code user supposed to deduce from this result? One must be <u>very</u> cautious in interpreting hydrogen combustion results from lumped parameter codes, because burning the hydrogen each and every time that combustible conditions exist - which is precisely what MELCOR does - is the functional equivalent of assuming that an ignition source is always present, when in fact there may be no ignition source.

GASFLOW II allows the prediction of local, heterogeneous gas behaviour during severe accidents - something which lumped parameter codes such as MELCOR (which assume homogeneous conditions within each calculation node) are incapable of doing. The consequences of deflagration at any given time can then be modelled with a code appropriate to the conditions at the time (e.g., a code which can model DDT, detonation, etc.).

The actual calculation of hydrogen combustion was not performed based on the GASFLOW II modelling. Rather, the goal was to understand whether, based on 3D CFD modelling, conditions conducive to energetic hydrogen combustion modes could occur in the WWER 1000 design, and to gain an understanding of where such conditions could develop and over what time period they might persist.

Two GASFLOW II calculations were performed for the ’in-vessel’ portion of the accident. This was considered to be the best use of the available resources and time. Extended calculations covering the full duration of the accident were not possible within the budget and time duration available for the calculations. It should be noted that the in-vessel portion of the accident is associated with the highest hydrogen release rate to the containment and the formation of a sensitive cloud is most sensitive to hydrogen release rate rather than to the total quantity released.

The ’base case’ calculation used the hydrogen source to the containment from the WWER 1000 small LOCA sequence calculated with MELCOR 1.8.5. The base case showed a peak hydrogen release rate of 0,2 kg/s into the steam generator box containing the pressuriser.

Another "sensitivity" case was also performed, using a synthetic hydrogen source from another design. In this case, the hydrogen was released from a primary loop in the other steam generator box (i.e., the SG box without the pressuriser opening to the upper containment). This was done to investigate whether this SG box could be more sensitive to hydrogen mixtures, and also to understand whether a larger hydrogen release rate (a peak rate of 0,8 kg/s) could yield sensitive mixtures over a larger volume and longer time period.

The base case calculation was based on the MELCOR 1.8.5 prediction of steam and hydrogen source rates. The result was that a very small, transient sensitive hydrogen cloud was formed in the immediate vicinity of the release point from the primary system, but that the sensitive mixture rapidly dissipated and no large, sensitive hydrogen cloud formed in the SG box. One interesting result is that over the time frame covered by the calculations, the passive autocatalytic recombiners (PARs) in the lower part of the containment do not "see" the hydrogen. Thus, recombination rates are quite limited for these recombiners. In addition, as the peak hydrogen release rate extends for only limited periods of time (of the order of 500÷1000 seconds), even the PARs, which do "see" the hydrogen don't have much of an effect in terms of depleting the hydrogen cloud of its inventory.

At the time this calculation was done, only the MELCOR 1.8.5 results of source rates were available. Other more detailed, mechanistic code calculations were being done with the SCDAP-RELAP code in order to ’qualify’ the MELCOR code results, but these calculations were not available at the time. Once the results were obtained from the first GASFLOW calculation, it was decided to run the code again with a higher hydrogen release rate. As this release source was from a different type of reactor, it was deemed to be a ’synthetic’ hydrogen source and the calculation was deemed to be a sensitivity calculation. This synthetic hydrogen source had a peak hydrogen release rate about 4 times higher than the base case.

In the sensitivity case, a rather large sensitive hydrogen cloud formed and persisted for several hundred seconds. The cloud occupied the upper portion of the SG box, and extended into one portion of the containment dome near the release point from the SG box to the containment annulus (which communicates between the upper and lower containment around the inner edge of the containment).

After these calculations were completed, the results of the SDCAP-RELAP calculation became available. These results indicate that the 0,8 kg/s release rate, which was used in the 'sensitivity case' using a synthetic hydrogen source, was not as conservative as previously believed. The SCDAP-RELAP calculation showed several periods in which peak hydrogen release rates exceeded 0,8 kg/s. Accordingly, the 0,8 kg/s release rate must be regarded as at least plausible since it is based on mechanistic code calculations for the WWER 1000 reactor coolant system.

The obtained results from the GASFLOW II calculations need to be understood in proper perspective:

- The calculations have considerable uncertainty associated with them.
- The calculations did not include modelling combustion of the obtained sensitive hydrogen cloud.
- The calculations were for a generic WWER 1000 and were not entirely specific to Temelín due to lack of access to detailed plant documentation and physical access to the plant to confirm geometric details.

4.2.3.2.4 Implications - The GASFLOW II Results in Perspective

The small LOCA sequence was selected for modelling because it was recognized that the containment atmosphere would be potentially conducive to hydrogen combustion (i.e., it would not be steam inerted), and because the hydrogen could be released into the steam generator boxes, which present a relatively confined release area that does not communicate vertically with the containment dome. The conditions in other types of sequences would be different.

If the release point is from the barbotage tank, for example, as it would be in most transient sequences, the hydrogen would not be trapped in the SG box but would instead communicate easily with the upper containment and be dispersed in the greater volume of the containment. In large LOCA core melt sequences, the in-vessel hydrogen release is smaller and the release rate is lower because most of the water in the primary system is lost out the break and is not available to produce steam in-vessel to react with the fuel cladding and produce hydrogen.

For the ex-vessel portion of accidents, in most cases (as a result of the introduction of a design change discussed at the Specialist Workshop), the reactor cavity door would be open and the core debris would spread over a larger area. The hydrogen release rate would (except for the initial phase of MCCI) be lower than for the peak in-vessel period, and the release would be to a less confined area of the containment, which communicates easily with the upper containment along the periphery of the containment. Thus, formation of sensitive hydrogen clouds would be much less likely for the ex-vessel portion of severe accidents in the VVER-1000/320 configuration.

The contribution of small LOCA sequences to the risk profile of Temelín then becomes important. It was identified during the Specialist Workshop that a very conservative initiating event frequency was used for small LOCA, which is about a factor of ten higher than other PSA studies. If a more typical small LOCA initiating event frequency were to be used, the contribution of small LOCA sequences to core damage frequency (CDF) in the updated Temelín PSA would drop from about $3,3\times10^{-6}$/a (representing 22,1% of CDF) to about $3,3\times10^{-7}$/a (representing 2,7% of CDF). At such a low CDF, in order for early containment failure to result from energetic hydrogen combustion to pose a significant hazard to Austria, there would have to be a very high conditional probability of all of the following conditions:

- A sensitive cloud would have to form. Whether a sensitive cloud forms in the SG box depends on the location of the small LOCA along the primary piping (not all small LOCAs are alike in their potential to form a sensitive cloud because some locations allow much more ready communication of the released cloud with the upper containment, and thus easier mixing with the larger containment volumes).

- The sensitive cloud would have to include a sufficient amount of hydrogen and be in an unfavourable geometry at the time of ignition (such that a detonation capable of threatening containment integrity could result). Not all small LOCAs have the same potential in this regard (small LOCAs range in size from 20÷50 mm effective diameter, and could in principle occur anywhere along the primary coolant piping and attached pipes).
- An ignition source would have to be available during the time that the sensitive cloud persists in the required concentration and quantity of hydrogen and in the specific unfavourable geometry. The presence of ignition sources during the transient period when a sensitive cloud is present is difficult to predict, however it should be noted that the PARs themselves could become ignition sources (Fischer 2003). (The precise locations of the PARs are not known to us (although the compartments in which they are located are known), and our calculations have used locations established by expert judgment rather than actual design information.)
- And even with all of these conditions, the cloud would have to undergo flame acceleration and DDT or direct detonation. It is important to understand that sensitive clouds do not *always* detonate – even with an ignition source present. Deflagrations or series of deflagrations can also be an outcome. This has been well demonstrated by experiments.

In the end, the GASFLOW calculation has showed that the MELCOR code can miss at least some conditions in which a sensitive (i.e., potentially detonable) cloud can form (the MELCOR calculation itself identified only marginally flammable conditions and no detonation potential was identified). Neither the MELCOR nor GASFLOW codes (nor other codes of which we are aware), however, have probabilistic models for combustion allowing for the possible presence or absence of an ignition source and the possible occurrence or non-occurrence of a detonation.

In other types of VVER-1000/320 severe accident sequences than small LOCAs, the potential for formation and ignition of a sensitive cloud *appears* to be less than in the small LOCA sequence, however this is an expert judgment rather than the result of a series of extensive calculations. The formation of a sensitive cloud (i.e., a cloud containing hydrogen which could upon ignition undergo flame acceleration and deflagration-to-detonation transition) as a result of a severe accident in a WWER 1000 appears as a result of the GASFLOW calculation to be physically possible, but its likelihood is uncertain. The specific sequence modelled has a low absolute frequency (estimated to be of the order of 3×10^{-7} per year), but this is not the only potentially affected sequence.

4.2.4 LB LOCA Sequences

The case of LB LOCA has very small probability of occurrence, made even smaller in view of the ‘Leak Before Break’ strategy implemented in most NPPs nowadays. In fact, according to the estimates provided in the 2002 Temelín PSA level 1, the frequency of core damage due to LB LOCA is about 3E-8/year, i.e. 100 times less than the frequency of core damage due to medium primary to secondary leakages. In addition the radiological consequences of LB LOCA are much less than those of PRISE severe accidents due to the protection provided by containment in case of LOCA events. Nevertheless, LB LOCA accidents are considered as the sequences with the fastest development of core damage processes and the highest decay heat releases at the time of RPV failure.

Within PN7 project LB LOCA sequences have been studied, and two cases – with molten corium contained within the reactor cavity and with molten corium spreading – have been analyzed. In the latter case the calculations were repeated after the Prague Workshop using the base mat concrete compositions provided by the Czech side and the updated PAR capacity. The main area of interest for this analysis was the challenge of the combustible gases generated to the integrity of the containment. The scenario was chosen because of its duration so that to allow estimation of the oxygen depletion due to operation of the PARs in the containment.

The scenario was assumed to start with a break in the cold leg 200 mm ED (break in the surge line to the pressuriser) with EFW available, HPIS and LIPS not available. To study possible effects of operator errors it was assumed that containment spray system CSS is recovered and actuated at the most challenging containment conditions with and without hydrogen deflagration. This made it possible to evaluate the hazards to containment integrity in case of unplanned hydrogen burn at the worst possible conditions (calculated with MELCOR code).

To study the effectiveness of the strategy of base mat protection against penetration, analyses were made for the case with corium contained inside the reactor cavity and with corium spreading outside cavity over 12 m^2 and 100 m^2.

The calculations showed that in the case of LB LOCA the penetration of RPV bottom head occurs at 4,21 h, with ejection of about 145 tons of corium and start of MCCI in the reactor cavity. In the case of molten corium remaining within the reactor cavity (steel door closed) the upper concrete layer (serpentinite) is ablated within 14 hours after the accident and the corium starts interaction with the base concrete.

The mass of gases produced during MCCI is high and for hydrogen it exceeds the amount of H2 produced during zirconium oxidation in the core. The mass of hydrogen produced during MCCI in the first concrete layer – serpentinite concrete is about 1100 kg, and in the lower layer of base concrete about 1900 kg. Duration for the base concrete ablation is about 33,6 h. The total time needed for penetration of the base mat is about 43 h.

In the case of corium spread out of reactor cavity the calculations confirmed that corium spreading extends considerably the time till baseman penetration by more than 26 h, so that the baseman is penetrated after about 74 h since the start of the accident.

For the case with spreading of the corium outside cavity the corium stratifies into 3 layers - metal layer (MET), heavy oxide layer (HOX) and light oxides (LOX) but very fast transformation takes place and practically from the onset MCCI is with 2 layers - metal and light oxides. According to the predictions of CORCON code (part of MELCOR) after HOX disappears MCCI becomes more vigorous and intensifies the generation of gases. Therefore this case with large spreading area leads to rather vigorous MCCI. The corium interaction with the base concrete produces 2940 kg of H_2 Long term pressure inside containment reaches about 0,55 MPa.

The temperature of the “cavity” compartment is rather low – about 560 K. The temperature of the rest of the containment compartments is around 480 K. The containment pressure increases quite fast and 45 hours after the accident reaches 4,5 bar. The peak recombination rate of 1 PAR for H_2 is about 0,18 g/s, and for CO – 0,19 g/s. The total recombined masses of H_2 and CO by 1 PAR during the calculated time period of 43 hours are 22,2 kg of H_2 and 20,8 kg of CO.

During the first 42 hours the atmosphere is not inerted. The H_2 concentration increases above 10 % at time 7,81 h. The amount of oxygen inside the containment is sufficient for deflagration (above 5 %). Therefore at this time deflagration of hydrogen is possible in the containment dome. The depletion of O_2 below 5 % due to the operation of the recombiners takes place 32,3 h after the accident.

The total amount of H_2 generated (both in-vessel Zr oxidation and from MCCI) is 3240 kg. Of this amount only 518 kg, representing about 16 %, is recombined by the PARs. The rest stays inside the containment posing a threat to the containment integrity in case of deflagration. Without deflagration the maximal H_2 concentration reached is about 19 vol.%. First conditions for deflagration appear in the ECCS compartment. Here again we can observe lower steam concentration in these compartments due to larger amount of surfaces per unit free volume and more extensive condensation in these volumes.

The first deflagration takes place at time 7,2 h in the ECCS compartment. The deflagration propagates to the annular corridor, to the SG compartments and to the containment dome.

The resulting peak pressure is 4,79 bar. A second deflagration follows at 10,5 h. In this case the initiation of the burning is in the “cavity”, which for this analysis is part of the annular corridor. Then the deflagration propagates to the SG boxes, to the annular corridor and the containment dome. The second pressure peak is higher – 5,55 Bar, but still it does not exceed the design pressure of the containment. The peak temperature reached as a result of the deflagrations is about 1000 ^{O}C.

After the second deflagration the amount of O_2 is below 1,18 %. Thereafter the recombiners slowly deplete the oxygen. The hydrogen concentration increases due to MCCI and is around 12 vol.%. The steam concentration stabilizes around 57-58 vol.%.

If the steam in the containment is condensed, the concentration of oxygen can increase above 5 vol. %. In the calculations it was assumed that the CSS was started, though the pressure inside the containment does not require such action. It was found that the operation of the spray increases insignificantly the O_2 concentration, but no deflagration follows and there is no threat to the containment integrity.

The general conclusion from the analyses is that the hydrogen and oxygen concentrations may pose a challenge to the containments of Temelín NPP, but the situation is far from dramatic as the peak pressure remained within the limits below the design pressure of the containment. It can be noted that the operator is instructed not to actuate CSS at the time when atmosphere is not inerted and wait with spraying until the hydrogen and oxygen concentrations become lowered by action of recombiners below flammability limits. The Czech calculations recognize that the case of an erroneous actuation of sprays the loss of containment integrity would be possible. They stress that the operators are well trained to avoid such errors, and the likelihood of such a scenario is very low.

On the other hand if the RPV fails and the molten corium is released to the reactor cavity, the hazard of basemat penetration is significant.

The calculations with MELCOR code showed that assuming implementation of SAMGs the containment integrity is not threatened by increases of pressure due to long-term gas generation or hydrogen burns. The main hazard consists in the possibility of containment baseman penetration.

4.3 Discussion of Phenomenological Issues

In the course of severe accident sequences there are a number of complicated phenomena that are taken into account. In the early phase of nuclear power development the processes involved were not exactly known, so that rather pessimistic assumptions used to be taken on their probabilities and consequences. US NRC identified them as “Unresolved Safety Issues” (USI) and a large programme of experimental and analytical work was undertaken to clarify and resolve them. Presently most of the USIs connected with severe accidents have been resolved, but some of them are still being studied on international level. The phenomenological issues of most importance for Temelín NPP safety are briefly presented below.

4.3.1 Vessel Failure

The consequences of Reactor Pressure Vessel failure depend very much on the pressure inside the RCS at the moment of vessel melt-through. If the pressure inside the RCS could not have been decreased below 1-2 MPa before the vessel break, the sequence of events is called “high pressure scenario”, otherwise “low pressure scenario” occurs. The scenarios of severe accidents are divided accordingly into the two following classes

- High pressure scenarios, including total blackout, total loss of feed water, small break LOCA with failure to depressurize the RCS in time before RPV break
- Low pressure scenarios, including LB LOCA, large Primary to Secondary leakage, and those severe accident scenarios where a successful RCS depressurization is achieved.

Each class of scenarios has its own specific features and the results are plant specific.

4.3.1.1 High Pressure

The results of German Risk Study Phase B [GRS 89] indicated that without accident management measures 98% of severe accidents would lead to high-pressure (HP) core melt. In high-pressure core melt scenarios the delivery of cooling water to the core is more difficult or impossible, and the reactor pressure vessel (RPV) rupture results in high pressure melt ejection (HPME). The ejected materials are likely to be dispersed out of reactor cavity into surrounding containment volumes as small particles, quickly transferring thermal energy to the containment atmosphere. In addition, metallic components of the sprayed core debris, mostly zirconium and steel, can react with oxygen and steam in the atmosphere, raising a large quantity of chemical energy that can further heat up and pressurize the containment. The term “direct containment heating (DCH)” is used as a summary description of the involved physical and chemical processes.

The magnitude of the containment loading that could be caused by HPME/DCH depends on various features of the plant design, especially the design of the reactor cavity, and also the availability of flow paths to the upper regions of the containment. Recent containment analyses have shown that, for a PWR with a typical large dry containment, DCH is only a threat if a large quantity of entrained debris (typically >50 % of the core inventory) is involved. The implicit assumptions made in these analyses are that the entrained mass is finely fragmented and that flows are unimpeded.

In reality, there are several mitigating factors that would considerably limit this energy transfer. The key factors are fragmentation of ejected corium, its de-entrainment by structures and efficiency of chemical and thermal interactions. Recent analyses [Henry 91], [Werner 94], [Ang 95], [Morozov 03] have all concluded that large dry containments are less vulnerable to the loadings from DCH than previously suggested by scoping containment analysis and the containment failure probabilities are small.

Another process can that was considered a hazard to containment integrity is the possible steam explosion due to molten corium interaction with water. The steam explosion loads on the containment were first considered in WASH-1400 and, because of the assumptions made about the nature of this event at that time, the failure of containment (due to in-vessel steam explosion generated missile) contributed a substantial fraction of the conditional probability for early containment failure. The work on steam explosions since that time led to more realistic estimates of the probability of containment failure due to in vessel steam explosions. The current estimation is that this conditional probability (i.e., given a core melt) is less than 0,001. [Seghal 96].

The hazard of ex-vessel steam explosion depends on the amount of water that is present in the reactor pit and the spreading compartment at the time of the RPV melt-through. There are many facets to the determination of the containment failure probability due to interaction of a corium jet with water in deep water pool, e.g. jet characteristics, the corium composition, the extent of fragmentation, the strength of the trigger required, the pressure pulse generated in the steam explosion and the fragility of the containment.

In addition to the pressure increases due to DCH and water evaporation the pressure increase due to simultaneous hydrogen burning should be considered. This issue is discussed in the Sections 4.3.2 and 5.5.

The ejection of molten corium under high pressure involves the hazards of molten corium attack against the containment liner and possibly against other surfaces involved in providing containment leak tightness. Such hazards should be evaluated and prevented as needed.

4.3.1.2 Low Pressure

The best protection against various threats listed above consists in bringing RCS pressure down before RPV failure. It is generally agreed, that below 1 MPa no HPME phenomena are of importance.

Moreover, depressurization of the RCS brings important advantages much before the RPV break, because it facilitates using various water sources to inject into the RCS and cool the core. For example, in phase A of German Risk Study where no AM measures to depressurize RCS were considered the core melt frequency was estimated as 10^{-4} [1/a], while in Phase B with depressurization strategies implemented the CDF was reduced to $2{,}6 \times 10^{-6}$ [1/a], of which HP scenarios constituted only $4{,}5 \times 10^{-6}$ [1/a] [GRS 89, p. 67]. Therefore, the intensive depressurization of RCS in case of severe accidents is generally accepted as one of the main SAM strategies.

4.3.2 Hydrogen Production, Distribution and Combustion

Controlling the release of large quantities of hydrogen during the course of a severe accident poses one of the greatest challenges for the design of NPPs. The main potential hydrogen source during a core heat up accident is due to the oxidation of zirconium fuel cladding and zirconium containing structural core materials with steam. The reaction of zirconium /steam is highly exothermic (587 kJ/mol) so that the reaction is self-accelerating once the fuel temperature has reached approximately 1100 °C. The sole limitation is given by the amount of available steam. This fact explains that very large quantities of hydrogen can be generated in short time spans, being eventually limited by steam starvation, although the core is still heating up. At the time of core slumping into the water of the RPV lower head there will be another short period of intensive hydrogen generation.

The hydrogen produced during in-vessel phase can be released from the RCS through pressuriser PORV or safety valves in case of such scenarios as blackout of LOFW, and flow to the relief tank and then to the containment dome, or through the break in the RCS piping in case of SB LOCA. In the latter case the concentration of the hydrogen near the release point depends on the room geometry. In the case of WWERs with horizontal SGs and flat SG boxes, the hydrogen will initially stay in the boxes before flowing into the containment dome, so that its concentrations will be locally higher than in PWR containments with vertical SGs and high rooms containing steam generators.

Additional hydrogen is generated during Molten Corium-Concrete Interaction (MCCI) after RPV failure. The quantities of generated hydrogen and carbon monoxide have been studied in many experiments and the calculation results obtained by means of CORCON code used in MELCOR code package to describe MCCI are well comparable with the experimental data. The rates of ex-vessel hydrogen generation are generally lower than those typical for in-vessel processes.

In terms of the threat posed by hydrogen combustion to containment integrity, several different combustion regimes should be recognized. Hydrogen deflagration - simple burning, producing no shock loading - is not likely to pose a threat to containment integrity. Generally the pressure rise due to deflagration is smaller than the containment design pressure, and no dynamic (shock) loads are involved.

There are three more energetic hydrogen combustion modes to be considered: (a) flame acceleration, (b) deflagration-to-detonation transition (DDT), and (c) detonation. Flame acceleration represents an intermediate and transitional behaviour between deflagration and detonation. The resulting pressure loads are higher, but still not typically threatening to containment integrity for a large dry containment. Transition to detonation (DDT) is more of a threat to containment integrity because detonation involves shock loading on structures. DDT requires specific conditions in order to take place. Global detonation is also possible, but very unlikely since conditions conducive to its occurrence are not normally encountered in large dry PWR containment plants.

It was recognized by the late 1990s that the WWER 1000 design might be more at risk from energetic hydrogen combustion modes due to structural layout considerations, especially the potential for DDT in the lower part of the containment [Kujal 97]. Since such hydrogen combustion modes are not modelled in the MELCOR code (or other similar lumped parameter codes), in PN7 this issue was addressed by the use of a three-dimensional CFD code (GASFLOW II) to identify the extent to which conditions conducive to DDT might be present in the WWER 1000 design.

Hydrogen hazards in large dry containments are primarily mitigated by the large free volume and high structural strength (compared with other designs) of the containment. Hydrogen combustion is also prevented under situations in which the containment atmosphere contains sufficient steam to suppress combustion. (Ignition is not possible with a steam concentration of 55% or higher, a condition which is often encountered in severe accidents not involving containment bypass.). Moreover, the hazard of deflagration decreases with pressure and according to Czech estimates at high pressures the probability of deflagration with significant steam contents is negligible.

In addition, the hydrogen source in the containment atmosphere in severe accidents is depleted in the long term by twenty-two passive autocatalytic recombiners (PARs) installed in the containment. The specific model installed at Temelín is rather small (Siemens FR 90/1-150), and is intended for design basis accident conditions to maintain the hydrogen concentration in the containment at less than 4 vol.%. Under severe accident conditions these PAR units do over the long term deplete both the hydrogen and oxygen concentrations in the containment by catalytic recombination of hydrogen and oxygen to form water vapour, but in view of their small size it is a slow process. Higher capacity PARs would be more appropriate for severe accident conditions.

Finally, venting the containment atmosphere to the environment can reduce the concentration of hydrogen in the containment. The Temelín SAMGs provide for a filtered venting path, which is nominally intended to reduce containment pressure. However, actuation of this filtered venting pathway necessarily removes hydrogen from the containment (and also reduces the pressure in the containment which could be present if hydrogen combustion takes place, thus reducing the threat of overpressure failure of the containment). The effectiveness of venting for reducing hydrogen combustion hazards has been widely recognized within the EU [EUR 14037] and the US [NRC 03].

4.3.3 Core-Concrete Interactions and Base mat Penetration

Molten corium-concrete interaction (MCCI) results in concrete erosion, which decreases base mat thickness and can lead to complete base mat penetration by molten corium. The temperature of corium at the moment of RPV failure is about 2700 K, while the solidification temperature is 2173 K. Since at the initial moment of corium-concrete interaction the corium contains significant amount of metal and the temperature of concrete dissolution is higher than the temperature of metal-steam reaction, the release of combustible gases (H_2 and CO) occurs during all the period of corium-concrete interaction.

Thus, the process of MCCI gives rise to two hazards:

1. Containment pressure increase due to production of non-condensable gases, including hydrogen and carbon monoxide, which may eventually burn, thus increasing the containment atmosphere temperature and pressure.
2. Base mat penetration, with loss of leak tightness of containment envelope, mixing of hydrogen with air, which may result in deflagration, and fission product leakage outside containment.

Liquid corium will spread on the floor of the reactor cavity. The results of spreading experiments with prototypic corium melts at Siempelkamp and Forschungszentrum Karlsruhe indicate average melt thicknesses of about 2÷4 cm [Spengler 02]. This shows a strong influence of cooling on the real spreading processes. The temperature of the molten corium is kept up by residual heating. In the course of molten corium penetration into the concrete the temperature of corium decreases, its viscosity increases and the rate of reaction with concrete goes down. The thickness of crust grows due to cooling and phase changes. Small-scale experiments showed that the formation of this crust decreases effectiveness of molten corium cooling from above with water, but in larger scale experiments the crust was cracking and letting water get into more intimate contact with molten corium. The rate of the MCCI reaction depends also on the generation of gases from heated concrete and their influence on the heat transfer from corium to the concrete.

Generally, the phenomena occurring during MCCI are complex and not fully understood. Several experiments have been performed or are under way to gain better understanding of these processes. Large-scale tests are performed in Germany at Siempelkamp CARLA plant in the COMAS facility [Steinwarz 01] with the aim to model the phenomena related with the retardation of the flow, which may lead to the final arrest of the corium.

4.3.4 Long-Term Overpressure

In the long term the main reason for possible containment integrity failure would be inadequate heat removal from the containment, with temperature and pressure increase above the design values. As the containment is the main and final barrier against release of radioactive products to the environment, the protection of containment integrity is recognized as the ultimate objective of Severe Accident Management.

Energy and mass releases into the containment during a core melt accident result in pressure increase. If the means of heat removal from the containment should fail, the pressure in the containment would slowly increase and after several days – if there were still no heat removal – the containment could fail due to overpressure in several days time.

In case of MCCI large amounts of heat are generated. In parallel, corium-concrete reaction involves generation of non-condensable gases, which increase containment pressure. If the fraction of hydrogen in the containment is elevated and the containment is inerted due to high fraction of steam, actuation of containment spray system might lead to de-inerting the containment, increase of hydrogen volumetric concentration and hydrogen burn or even deflagration to detonation phenomenon. Therefore, the actuation of CSS would not be allowed in the late stage of the accident, and the pressure in the containment would steadily grow till the containment failure.

To prevent it, **filtered venting of the containment** has been proposed. Various methods of filtered venting of containment have been implemented in Sweden, France and Germany, and are being introduced in other countries. In Germany pressure relief system includes deep bed fibre filters and molecular screens for elemental iodine or venturi scrubber with retention capacity for aerosols 99,99%, and for elemental iodine 99%. In France pressure relief systems are provided with sand beds, assuring effective retention of volatile fission products.

Some NPP designs include **dedicated systems for containment cooling** by natural convection, which can assure effective heat removal even during total blackout conditions and prevent pressure build-up inside the containment.

5 EVALUATION OF ACCIDENT MANAGEMENT STRATEGIES

5.1 Prevention and Mitigation of PRISE

The problems of prevention and mitigation of PRISE sequences are given the highest attention in all NPP analyses, which also has been reflected in the number and detail of VLIs formulated in the PN7 project on the subject of PRISE accidents.

VLI No.	VLI title / description
Prevention	
8.1.1	Have there been any cases of primary collector breaks in those SGs that were made using rolling technology of fastening of heat transfer tubes in the collectors?
8.1.2	Have there been any cases of deviations from secondary circuit chemistry observed during operation of Temelín so far?
8.1.3	Are the temperatures at water-steam interface kept outside the temperature range of phase transition for the steel from which the primary collectors are made?
8.1.4	Is the N16 system installed and observed on-line by the operator?
8.1.5	Are the criteria for tube plugging established? Are they the same as used in Dukovany NPP?
8.1.6	Are the eddy current methods implemented for detection of developing tube cracks?
8.1.7	Have the analyses of possible cracking of SG tubes been made for the case of DBAs with EFWS operation and for cases of severe accidents with High Pressure scenarios?
8.1.8	Have the SAMs involving cold-water injection into SGs during severe accidents been analyzed from the standpoint of possible tube breaks?
8.1.9	Has the actuation point for BRUA forced opening for RCS cooling been analyzed from the standpoint of possible breaks in the SG tubes due to tubes uncovery?
Mitigation	
8.2.1	Are the Fast Closing Main Steam Isolation Valves qualified for water hammer and steam-water or solid water flows?
8.2.2	Are the lines to BRUA and safety valves qualified for steam – water mixture and water flows?
8.2.3	Is the operator allowed to throttle HPIS pumps? When?
8.2.4	Are there any means for providing water to SGs in case of EFWS failure?
8.2.5	Is there in SAMGs the requirement to keep SG filled with water?
8.2.6	Is RCS pressure reduction through PRZ spray injection possible in case of LOOP?
8.2.7	Is PRZ spray injection protected against single failure?
8.2.8	Are there any means to provide firewater or other water supplies to the ECCS tanks?
8.2.9	Is forced cooling of RCS foreseen in case of PRISE in SAMGs? What is the entry point?
8.2.10	Is high boron concentration in primary coolant required before intensive RCS cooling can be started? How and when can it be achieved?
8.2.11	Does depressurization of the RCS result in leakage stopping before loss of primary coolant and core uncovery? For what entry point and bleeding capacity?

State-of-the-art requirements and practices

There is a broad international consensus in OECD countries on the necessity to improve the design of NPPs in order to avoid containment bypass sequences. It has been observed that the Postulated Initiating Event (PIE) of Steam Generator Tube Rupture (SGTR) is for some designs still classified in a category of PIEs "not expected to occur". Given that the operating experience has proven that the occurrence of this PIE is not so low, the TSO Group stated that the categorisation of this PIE has to be reconsidered [TSO 01]. According to the opinion of GPR/RSK and European utilities sequences with SGTR should be considered assuming simultaneously a failure to close of the main steam isolation valve or steam relief valve.

The importance of PRISE is highlighted by Westinghouse SAMGs, in which the highest priority have actions that aim at mitigating radiological releases after PRISE and preventing SGTR during high pressure accident sequences such as blackout or SB LOCA.

Primary collector rupture was not included among the Design Basis Accidents in the original design of WWER 1000 NPPs. After cracks were revealed in a number of steam generator collectors in various WWER 1000 NPPs, close attention was paid to this hazard. It was found that one of the main reasons of cracks was explosive bonding, which involved excessive stresses in the collector. Another factor influencing the process of collector cracking is the secondary side chemistry. Strict adherence to the requirements for secondary water chemistry is important for reducing the rate of crack development. However, although the requirements in this respect have been known for a long time, many NPPs in Russia and Ukraine observed cracks in their SG collectors due to deviations from the required chemistry.

The injection of cold water from EFWS on uncovered SG tubes can result in cracking of hot tubes. The issue is important in the implementation of SAM measures, especially in the case of high-pressure scenarios. Both EOPs and SAMGs should be analyzed and the actuation points for EFWS injection chosen in such a way as to minimize the danger of SGTR.

All valves exposed to possible steam-water mixture flows after PRISE should be qualified for such flows, or remain closed during the transient. If BRU-A is opened during the initial blowdown and fails open, then the leak can turn into a severe accident with potentially very large radioactive releases.

RCS pressure should be reduced in case of PRISE so as to reduce radioactive leaks from the primary side and – what is even more important – reduce losses of primary coolant inventory. One of standard measures for RCS pressure reduction is injection through PRZ spray system. However, in the case of LOOP and loss of RCPs the pressure head on the RCP delivery side disappears and the sprays are not functional. In some NPPs PRZ sprays can be provided with water from the RCS make up system supplied by Diesel generators. In some WWER NPPs it has been found, that even if make up system can inject into the PRZ spray line, that line is prone to single failure, because in one section between the make-up system and the PRZ there is a single valve which can fail closed. It is worth noting that in Loviisa the system was redesigned and parallel lines to PRZ sprays were installed to keep this mode of RCS depressurization in case of that valve failure.

The operator should be allowed to throttle ECCS injection if PRISE is identified. All sources of available water should be used in case of need to keep the core covered. Appropriate SAM strategies should be developed.

Current plant status

Temelín NPP is fully aware of the hazards involved in PRISE leakages and various preventive measures have been implemented to limit the frequency of PRISE. These include the improved technological method of hydraulic expansion to fasten SG tubes in the primary collectors instead of explosive bonding. According to international WWER experience, there have been no cases of cracking collectors produced using this technology. On-line monitoring of secondary side chemistry has been implemented. The turbine condenser tubes produced originally from stainless steel were replaced with tubes made of titanium to ensure leak tightness and prevent the leakage of cooling water into the secondary circuit. Owing to the change of tube material it was possible to increase pH of feed water to above the value of 9. Each SG at Temelín NPP is equipped with chemical diagnostics that enables taking of samples from water volume on the secondary side and with automatic monitoring of blowdown water. The operator can observe on line the actual chemistry of water.

Advanced NDT methods are used in Temelín for ligament checking. To detect SG tube failures, which are still possible on line monitoring for SG integrity is realized by means of gamma scanning of SG blowdown water. An extensive programme of SG integrity assessment and maintenance has been implemented.

N16 detection system to detect early signs of leaks is installed and in operation. The system is not sufficient per se to protect against developing leakages, but it helps in early detection of leaks and it also serves for more accurate identification of the affected SG if the leak does develop. The criteria for tube preventive plugging before large leaks appear are established and eddy current methods are used for early detection of tube cracks. The analyses of possible cracking of SG tubes in high-pressure scenarios have been made and SAMG strategies include keeping water level in SG high enough to maintain SG tubes covered with water.

The design of the collector header cover in Temelín was modified to reduce consequences of header lifting due to connecting bolt rupture, reducing the break flow area from 100 mm equivalent diameter down to a value equivalent to 40 mm diameter leak. All these measures have reduced the frequency of PRISE initiating event in Temelín NPP, which is reflected by much reduced likelihood of core damage due to medium size PRISE, from 4,3E-5 to 3,09E-6.

The probability of PRISE induced in the course of an accident is reduced by proper strategies introduced in the plant EOPs and SAMGs. They include keeping the SG secondary side filled with water to maintain integrity of SG tubing. There are also developed emergency means of water delivery to the secondary side of SGs, which can be applied even if EFWS fails, such as providing firewater connections and a reserve diesel generator that can be connected to any Temelín unit.

There are several possibilities to mitigate PRISE if it does occur, starting with RCS pressure reduction by injection to the pressuriser. The injection is provided by make-up system, which is powered by diesel generator and designed against single failure criterion so that parallel control valves are provided in the injection lines. The SAM strategies allow operator to throttle ECCS injection if PRISE event is identified. This would be achieved by gradual switching off the individual HPIS trains and the use of make-up system, which has capabilities for throttling the make-up flow. To replenish the water inventory in the RCS the operator can use two large pools, which are usually empty, but can be filled with water for fuel reloading purposes. During an accident these pools will be filled up with water and made available for refilling the RCS if the inventory of the RCS and ECCS is low.

The question of major importance is the qualification of BRU-A for water-steam mixture flow. This issue is considered in another project under the Melk Process (PN3). However, even if the BRU-As are shown to be qualified for two-phase flow and water flow, in case of multiple demands during the accident they can fail as discussed in Section 4.2.1. The piping lines conducting to BRU-As have been calculated and their strength was found to sustain dynamic loads due to water-steam or solid water flow conditions.

On the secondary side there are several possibilities of refilling the steam generators, one of them being the fire water system. The strategies for this purpose have been developed, the sources of water are plenty and all of them are specified in SAMG. In addition there is a large time margin to implement these measures. It was shown in the calculations of the PRISE scenario in which a core meltdown due to lack of water on the secondary side of steam generators occurs after 5 days. Taking into account the numerous sources of water available to feed the secondary side it is incredible to assume that any team of operators fail to use these sources during 5 days to fill up the secondary side of steam generators.

The Austrian side did not find out the numerical values for entry point bleeding capacity and other parameters used in WOG SAMGs, but the general statement was that SAMGs are entered if the operator finds that the EOPs are unsuccessful. The criteria will be introduced in the symptom oriented (SO) EOPs when the plant is ready for the implementation of SAMGs.

Given the long time period available in which to accomplish primary makeup (so long as the BRU-A valves continue to function properly), it is nearly inconceivable that the operating staff (together with the accident management team in the TSC and all available regulatory and industry technical support) would fail to provide primary makeup within 5 days to avoid core damage. This outcome generally indicates the robustness of the design and the EOPs for PRISE accidents.

If a BRU-A valve sticks open in a PRISE accident, the scenario then proceeds much more rapidly, with core melt and vessel failure within 18÷19 hours. Even in this case there is considerable time available to accomplish primary makeup, including potential for recovery of the situation by the second shift to come on duty since in an 18÷19 hour time period at least one shift change will take place. Thus, in the extreme case in which a particular mindset develops that results in depressurisation and primary makeup not occurring during the shift, in which the accident has started, continuation of such a mindset in a new crew of operations and TSC staff coming on duty is less likely.

Even if a PRISE accident proceeds to core melt, it is likely as a result of EOP and SAMG provisions adopted or pending adoption that the affected steam generator will continue to receive feedwater throughout the duration of the accident. This will result in a reduction of the resulting "source term" of radioactive materials released to the environment (a mitigation strategy).

The MELCOR calculations performed in PN7 for the PRISE accident indicate that the scenario defined in the PSA is very likely not to develop into a severe accident due to the extended (5-day) time frame within which the accident is recovered using water sources available onsite. Thus, the updated PSA core damage frequency represents an over-estimate. The medium PRISE accident (40 mm D_{eff}) only realistically goes to core damage if the BRU-A sticks open. Above we estimated a 10% chance of this happening due to repeated demands on the BRU-A to open to relieve steam pressure in the PRISE sequence. If the BRU-A valve were to fail closed, a similar situation would be posed to the main steam safety valves, which would then themselves be subjected to repeated demands to open.

The updated PSA estimates the frequency of medium PRISE severe accidents to be $3{,}09 \times 10^{-6}$ [1/a]. The MELCOR calculation results combined with a preliminary assessment of the likelihood of the BRU-A valve sticking open indicate that the CDF should be reduced by a factor of ten to $3{,}09 \times 10^{-7}$ [1/a] (assuming that the BRU-A is qualified for water and two-phase flow). In the small LOCA discussion in Section 4.2.3.2, the CDF due to internal events was reduced from $1{,}49 \times 10^{-5}$ [1/a] to $1{,}19 \times 10^{-5}$ [1/a]. When considering the reduction estimated above for the PRISE accident, the total CDF due to internal events could be further reduced to $9{,}1 \times 10^{-6}$ [1/a].

Evaluation

The hazards involved in PRISE accidents are well recognized, the strategies appropriately developed and the technical means are provided to cope with PRISE events. The strategies adopted have resulted in a reduction in both the likelihood and consequences of most PRISE accidents. The PSA results show a reduction from $4{,}3\times10^{-5}$ [1/a] to $3{,}0\times10^{-7}$ [1/a] for Medium PRISE accidents (some of the reduction comes from better understanding of the thermal-hydraulics and success criteria for the accident, and some comes from optimized EOP and SAMG actions). Reduction of consequences arises from the SAMG strategy of maintaining water in the affected SG to 'scrub' fission product releases in the event that a PRISE accident proceeds to core damage.

5.2 Prevention and Mitigation of Station Blackout

As the main feature of this scenario is the loss of power, the VLIs below address mainly the aspects connected with the availability of power sources to prevent and later on to mitigate the accident.

VLI No.	VLI title / description
8.3.1	Are there means to provide off-site power from a reliable source besides the main power network? (e.g. from a hydro power plant, as in Dukovany)?
8.3.2	Are there means to provide cross-connection to the other unit on the site?
8.3.3	Are there any mobile diesel generators?
8.3.4	Has the capacity of batteries been extended to or beyond 2 hours?
8.3.5	Is it possible to operate (open, close) PORV, BRUA and other valves needed for SAM implementation after loss of power lasting longer than 2 hours?
8.3.6	Is there diversity provided in Diesel generator power supply?
8.3.7	Other than main feedwater, auxiliary feedwater, and emergency feedwater, are there any other capabilities to inject water into the steam generators?

State-of-the-art requirements and practices

The recovery of electric power is the essential condition of success in prevention of core damage or later on in prevention of the basemat penetration in the case of severe accidents. The calculations performed within TACIS programme for WWER 1000 NPPs indicated that the recovery of electrical power within about 2,5 hours after station blackout makes it possible to avoid any significant core damage, and with the recovery within 3 hours it is possible to stop the process of core melting [TACIS 02]. Even after RPV failure the recovery of electric power makes it possible to actuate LPIS and inject water onto the molten corium pool, providing top cooling of the corium.

The problem of battery capacity is more complex. The original design capacity of batteries in WWER 1000 units was low, below 1 hour. According to nowadays practice, 2 hours are required. However, in many severe accident sequences the time needed to get the plant to stable steady state is longer. Such valves as PORV, the valves of EGR system, or reliable service water system valves etc. must be operable much longer than 2 hours. The analysis of needs in this respect should be made and the necessary long-term operation of all equipment needed under severe accident conditions should be assured. Since diesel generators are of key importance for providing reliable power under accident conditions, they should be protected from common cause failures. In some NPPs diversity within the DG system is provided.

Current plant status

The hazard of station blackout is reduced by the variety of sources of electric power available, including the power from two external lines, from the other unit at Temelín NPP, from 3-train emergency DG system and from two additional non-safety class diesel generators. The non-safety diesels are of the same design as safety diesels, but without some protective circuitry required of safety diesels. Thus there is some potential for common cause failures of diesel generators.

In the initial PSA study (1996) a battery depletion time of one hour was assumed. In the updated PSA a battery depletion time of 2÷3 hours is used [Mlady 03a]. Temelín staff informed us that actually the capacity of batteries is greater than required by design and with proper load management provides electric power for 2 to 4 hours after the accident [Sykora 03]. This would provide possibilities to open or close the valves needed for SAMG implementation. However, the capacity of batteries remains a weak point. It is worth noting that the example of some other WWER NPPs shows that the batteries can be and, in fact, have been exchanged in several plants for other units of higher capacity.

In case of blackout by definition no electric power is available and so there is no possibility to inject water to the core from any active systems. The injection from Safety Injection Tanks (SITs) can be achieved, but it is not sufficient to prevent core melting. Thus, station blackout would lead eventually to core melt and RPV failure

There are several means of injecting water to the secondary side of SGs, as mentioned in section 5.1. However, although there are 2 spare diesels, which can be connected to any safety bus at either unit, the occurrence of a station blackout means by definition that these 2 diesels are unavailable. The fire protection system pumps are motor driven and unavailable in SBO conditions; the same is true of essential and non-essential service water. The only remaining way at this point to get water to the secondary side of the SGs is by fire trucks. In order to be able to control the valves for this and other operations it is necessary to have batteries available in the long term.

Evaluation

The preventive measures at Temelín NPP correctly address the issue of station blackout.). Moreover, SAM strategies include measures to prolong battery lifetime by re-structuring the load profile much beyond the design period of 1 hour. Nevertheless, it might be desirable to exchange batteries or include in the system additional power sources that could provide electric power during the blackout conditions

5.3 RCS Depressurisation

VLI No.	VLI title / description
8.4.1	What is the design basis for high vent points in the primary system? What is the capacity of Emergency Gas Removal system lines?
8.4.2	Have there been analyses or experiments to establish real capacity of Emergency Gas Removal system?
8.4.3	Do analyses take credit for 30 kg/s or for limited 20 kg/s capacity of EGR?
8.4.4	Was the capacity of PORV to remain open at low pressures demonstrated? Is its reliability to be opened at low pressures equal to that of the PORV to open at high pressures?
8.4.5	Are the lines leading to BRUA and to PORV qualified for water-steam mixture and solid water flows?
8.4.6	Is parallel gas removal assumed from several points in EGR system and from PORV? Have there been analyses of physical restraints of steam flow in such cases? Were they submitted to SUJB and approved?
8.4.7	Was the weakening of steel at high temperatures taken into account in high-pressure scenarios and SG tube rupture analyzed?
8.4.8	Were the consequences of sudden recovery of EFWS and injection of cold water into the SG considered?
8.4.9	Were the consequences of high temperature failures of measuring equipment and the delays or errors in SAM implementation considered?
8.4.10	Is the formation of water seals in the RCS considered in the analyses?
8.4.11	Has the model of natural circulation in the core with bypass flow downwards been considered?
8.4.12	What are the means of water injection onto molten corium ex-vessel
8.4.13	Has the candling heat transfer coefficient for core melt established in CORA tests been considered?
8.4.14	Has the stratification of corium with possible RPV failure at the sidewall been taken into account?
8.4.15	Has there been an analysis of high temperature fuel coolability with water? Has the RPV weakening due to high temperatures been taken into account?
8.4.16	Does the depressurization strategy take into account the time during which the core was in high temperatures in choosing operator's actions?
8.4.17	Could you generally characterise the end pressures achievable, under a variety of accident scenarios, using these depressurisation capabilities?
8.4.18	What is your assessment of the final pressure (at the time of vessel failure), which is required in order to avoid HPME/DCH for Temelín?

State-of-the-art requirements and practices

The very importance of depressurization of the RCS is generally recognized, and according to both safety authorities and European TSOs [TSO 01] for future NPPs it is required „to transfer high pressure core melt sequences to low pressure core melt sequences with a high reliability“ [GRP/RSK 93]. The design provisions should include diverse secondary side heat removal systems to depressurize the primary in a controlled and reliable way, sufficient primary feed and bleed capacity in the event the secondary systems are lost, and adequately sized pressuriser relief devices or a specific dedicated system for direct primary depressurization as the last line of defence. The discharge capacity and the reliability of the depressurization system would have to be demonstrated for severe accident conditions. It is recommended that key valves be made as reliable as the valves used to prevent an over pressurization [EUR 20163].

In the operating NPPs the means available for RCS depressurization are not as diverse and reliable as those required of future NPPs. As long as there is water injection to the secondary side of steam generators, opening relief valves on the secondary side is an effective method of RCS depressurization. However, under severe accident conditions loss of the main feed water system, auxiliary feed water system and even emergency feed water system has to be postulated. Under such conditions forced opening of the secondary side relief valve to lower the secondary side pressure has been shown in PN7 MELCOR calculations to bring only temporary RCS pressure drop. In the long run this strategy can result in faster loss of secondary water inventory and faster core uncovery. It is recognized, that secondary side depressurization without availability of sources of water to inject to SGs is not sufficient per se to depressurize the RCS under severe accident conditions.

In the aftermath of TMI-2 accident, in which the presence of non-condensable gases hampered coolant circulation in the RCS, the requirement to install high-point vents was imposed in US plants. In WWER 1000 NPPs of 320 type the high vent points are installed on the top of the RPV, of the pressuriser and of each of the primary collectors in the steam generators. Since after gas removal to the pressuriser bubble tank and eventual burst of the bubble tank membrane the gases are released to the containment, the vents, which avoid that, are preferred for use as a means of depressurization of the RCS. The effectiveness of the system depends on the flow capacity.

TACIS calculations for WWER 1000 NPP showed that in case of maximum mass flow rate of 20 kg/s the Emergency Gas Removal System is not enough to assure reliable RCS depressurization under all severe accident conditions, while at 30 kg/s the RCS can be depressurized reliably before RPV failure [TACIS 02]. In Kozloduy NPP the tests at low-pressure difference and their extrapolation to high pressures suggest that the maximum EGR flow rate will be below 30 kg/s. Thus the question of reliability and capacity of EGRS lines is important.

If the EGRS is not sufficient to bring the plant to stable steady state conditions then the opening of PORV is necessary. In some WWER 1000 NPPs (e.g. in Balakovo) PORV cannot be opened below the operational RCS pressure [Morozov 03], but in most Western and Eastern NPPs PORV can be remotely opened at any pressure. The requirements regarding the reliable opening of PORV at low pressures are very high. In some NPPs that also have PORVs capable of being opened at low pressures (e.g. in Loviisa NPP), special dedicated depressurization systems have been designed and built.

In the process of depressurization the lines and valves of the secondary side of SGs, EGRS and PORV would be operated with water-steam mixture and solid water flows, while their design basis involves only steam flow. High mechanical stresses connected with water flows in those lines can result in their failure. In the case of the line leading to PORV this would correspond to SB LOCA and loss of control over the PORV opening and closing. In the case of the valves on the secondary side of SGs the consequences would be worse, because in case of PRISE a bypass of containment would result. Therefore, qualification of these lines for steam-water and water flows is necessary.

Current plant status

Temelín NPP realizes importance of depressurization. It is on a leading place among its SAMG strategies. The main technical means to achieve depressurization is to use the power operated relief valve (PORV) on the pressuriser or the emergency gas removal system (EGRS).

PORV is designed so that it can be remotely opened at low pressures and kept open. The minimum overpressure needed for its opening is 0,5 MPa. The design with an electromagnetic valve used as the pilot valve ensures reliable opening whenever the operator cuts the current to the electromagnetic valve. The loss of current to the pilot electromagnetic valve results in its opening; it is a passive arrangement, not subject to any failure of active elements. Once the electric current is lost, the pressure difference will open the valve. The value of 0,5 MPa overpressure is needed to overcome the force of a spring. Since the RCS pressure should be brought to about 1,0÷1,5 MPa in order to avoid HPME and DCH, the PORV can be maintained open throughout the whole period of RCs depressurization. Thus the opening is sufficiently reliable to fulfil the requirement of TSO that "the opening of PORV at low pressures should be as reliable as opening of a safety valve at high pressures" [TSO 01].

The lines leading to PORV have been checked to sustain dynamic loads due to possible steam-water mixture flows. The calculations made by ÚJV Řež for Temelín with RELAP code show that the opening of PORV is enough to assure timely depressurization of the RCS. The analyses performed within PN7 with MELCOR code confirm that. It should be noted that feed and bleed strategy is part of new symptom based EOPs and the calculations of depressurization with PORV were competently checked within the related process of EOP verification and validation.

The second line of defence is using EGRS. This system is designed to remove gases that can be accumulated in the high points of the RCS under accident conditions, in particular in the PRZ, RPV and primary collectors in the SGs. It can be also used for coolant removal from the RCS. According to Temelín staff, the calculations of Energoproekt Prague showed that EGRS is sufficient for RCS depressurization. This statement is based on calculations performed using RELAP code, which is a recognized tool for thermal hydraulic evaluations. On the other hand, according to PN7 calculations, and also according to Russian calculations within TACIS, the EGRS capacity is just at the limit needed for successful depressurization. If this system were the only means of pressure reduction in Temelín, the matter would need further consideration, but since the primary means of action is opening the PORV, this is sufficient.

In addition, Temelín NPP stated that the calculations of all scenarios would be repeated this year, using MELCOR 1.8.5 (so far only MELCOR 1.8.3 was used). After that Czech experts will review the question of effectiveness of depressurization again.

If the depressurization is not achieved fast enough, there might be a danger of consequential ruptures of the primary system pressure boundary within the steam generators with resulting PRISE and containment by-pass. The prevention of this failure is important for successful SAM. No information was obtained on the question of weakening of steel in high temperatures and possible thermal shock on SG tubing after EFWS recovery. However, the SAMG stress the necessity to keep the SG tubes covered with water, so that the conditional probability of accident induced tube break is maximally reduced. The formation of water seals was considered in the RCS model used in MELCOR. No information was obtained on the model of flow inside the core, the heat transfer coefficients used during fuel candling, corium stratification and the RPV weakening with high temperatures. The Czech side provided information about the special model of RPV bottom rupture that had been used in MELCOR calculations to reflect special WWER geometry.

It is worth to be noted that all these parameters will be determined again using the latest MELCOR code version 1.8.5 in the calculations to be performed by the end of 2003. This version of MELCOR has already been implemented in ÚJV Řež and is ready for use once the order of Temelín NPP is finalized. In version MELCOR 1.8.5 the features characteristic for WWER 1000 geometry can be fully modelled, including mechanistic determination of the bottom head rupture of the RPV.

Evaluation

The measures available in the plant are sufficient for timely depressurization of RCS. Temelín NPP has two lines of defence in this respect (PORV and EGRS), which is better than in many other NPPs of similar vintage. The WOG SAM strategies being implemented in the plant recognize the importance of depressurization.

The ETE-PSA estimate is that the probability of the PORV to fail is 2%, which is equivalent to the overall failure probability of the depressurization system.

The Specialist's Team would recommend the Austrian Governement to consider monitoring he calculations to be made by Temelín NPP with MELCOR 1.8.5 to see confirmation of capability of PORV to depressurize reliably the RCS. If its reliability at low pressure is lower than that announced during the Prague meeting, the credit taken for EGRS operation should be checked.

5.4 Containment Failure Prevention

5.4.1 DCH and other early threats to containment integrity

VLI No.	VLI title / description
8.5.1	Has DCH been considered? What was the amount of corium assumed to be dispersed in the containment?
8.5.2	Is reactor cavity well connected with the containment, or do the ventilation channels provide only limited flow cross section area to release steam from the reactor cavity to the containment after RPV failure?
8.5.3	After high pressure RPV failure, has the impact of molten corium against containment liner been considered and possible damages to the liner taken into account in determining containment leak tightness?
8.5.4	Has the impact of short-time high temperatures in the containment been considered while predicting that the containment integrity will be kept? Has a stress-temperature analysis been performed for ventilation system valves providing containment boundary?
8.5.5	Are there any reliable means of opening the door to the reactor cavity to facilitate corium spreading?
8.5.6	Has the danger been analyzed of the door being blown out by overpressure in the reactor cavity and hitting the containment wall?
8.1.10	What assessment has been made concerning the continued integrity of the spent fuel cask transfer hatch at elevation of the area outside the reactor cavity?
8.7.5	What is the physical and experimental basis for the conclusion that parallel hydrogen burning and DCH are not predicted for Temelín in case of HPME/DCH due to vessel failure at high pressure in a severe accident?

State-of-the-art requirements and practices

The worldwide-accepted state of the art practice is to demonstrate that the likelihood of early RPV-failure due to HPME as well as DCH as a consequence is very low.

Current plant status

Once Temelín has implemented the planned depressurization systems and procedures the plant status will be sufficient. Furthermore, calculations indicate that even in the case of HPME and subsequent DCH the containment failure pressure is not reached.

For the cavity door opening procedure it should be mentioned that, if a pressure scenario results in corium ejection and hot gases flow through that failed door opening the containment liner on the wall opposite the door opening can be damaged. It is also possible that the molten corium will be distributed non-uniformly causing non-uniform rates of penetration through the concrete.

Another hazard connected with HPME is due to the presence of a large metal hatch (opened during refuelling to transfer spent fuel casks to a rail car at elevation +0,0m) in the floor outside the reactor cavity, which serves as part of the containment boundary during normal operation. If HPME occurs, core debris could be deposited adjacent to and onto the hatch, degrade the seal between the hatch and the containment floor, or degrade the hatch itself, which might fail due to melting and/or a combination of thermal degradation and the ambient containment pressure.

Failure of the hatch cover would allow release of radioactive gases (as well as combustible gases such as hydrogen and carbon monoxide) and aerosols (and possibly core debris) outside the containment into the reactor building, which is not a hermetic structure. Czech side acknowledged these concerns during the Prague workshop and stated that it is planned to install removable walls that will prevent ejected corium from flowing outside a limited area, and in particular protect the containment liner and the spent fuel transfer hatch from contact with the molten corium.

Evaluation

Similarly to other NPPs with large dry containment, the hazards of early containment failure due to DCH in Temelín NPP have been evaluated as negligible and the strategy of depressurization included in SAM in Temelín reduces further such hazards. On the other hand, several technical measures are planned to be implemented in the plant, but have not been installed and no details of technical design have been disclosed.

Further monitoring is needed for the implementation of the mechanism for early opening of the steel door between the reactor cavity and the neighbouring equipment room and the installation of steel walls preventing ejected corium from flowing outside the designed area.

For monitoring the additional measures related to DCH prevention or mitigation, information is needed about the design requirements and the installation for the flow-deflecting wall (e.g. ability to withstand long term heat, flow impact and diversion under reduced pressure melt ejection etc. are of interest).

5.4.2 Basemat penetration

VLI No.	VLI title / description
8.6.1	Is it possible to flood effectively with ECCS water or with other water sources the molten corium pool in the reactor cavity or after spreading into the instrument room?
8.6.2	Can the reaction of molten corium with concrete be stopped before complete basemat penetration? Would power recovery be enough for it?
8.6.3	What is the minimum thickness of the basemat necessary to keep the weight of molten corium and the pressure in the containment?
8.6.4	Are there analyses to show the consequences of complete basemat penetration including possible hydrogen burning in the rooms below the basemat and fission product releases to environment?
8.6.5	How was the result reached quoted by [Sykora 01b], that with late ECCS recovery the depth of concrete ablation after 300 000 seconds is only 1.4 m? What mechanisms of water-cooling of corium were considered? Was credit taken for corium cooling with thin layer of water?
8.6.6	What is the composition and What are the dimensions of the concrete, including reinforcing material, in the containment floor between Elevation +10.8 meters and the floor level outside the reactor cavity?
8.6.7	What is the composition and what are the dimensions of the concrete, including reinforcing material, in the containment in the reactor cavity floor between Elevation +13.2 meters and the top of the reactor cavity floor?
8.6.7a	What is the composition and what are the dimensions of the concrete, including reinforcing material, in the containment in the cylinder wall between Elevation +10.8 meters and the top of the cylinder wall?
8.6.7b	What is the composition and what are the dimensions of the concrete, including reinforcing material, in the containment for the interior walls in the reactor cavity in the area above Elevation +13.2 meters?
8.6.8	There is a heavy steel door between the reactor cavity and the lower part of the containment. Do you envision operating Temelín with this door normally open? If not, what physical conditions (e.g., pressure, temperature, etc.) are required for force open the reactor cavity door if it is closed at the start of an accident and on what basis these conditions were determined (e.g., engineering judgement, stress calculations, etc.)?
8.6.9	Are there any means of opening the reactor cavity door without entering the containment? If not, under what conditions, if any, to do the Temelín SAMGs foresee a containment entry for the purpose of manual opening of the reactor cavity door? What are the instrumentation, alarm, phenomenological, or other cues to the operators to accomplish this action, and how much time is available from the occurrence of these cues in order to accomplish this action? From the time of entry into the outer-most door into the containment, until the action is completed and exit from the outer-most door from the containment is completed, how much time is required?
8.6.10	How do the Temelín SAMGs consider actions after RPV melt-through to protect the local compartments in that area? What guidance is available on actions that should be taken if this is not successful?

State-of-the-art requirements and practices

The problems of molten corium-concrete interaction were not given high attention in the design of NPPs of the vintage similar to Temelín (early 1980s), and in the reviews of NPP safety even in recent years. For example, in the IAEA TECDOC on Generic issues for NPPs with light water reactors [TECDOC 1044] the problem of basemat protection is not mentioned at all, even in the issue of "containment integrity during severe accidents", which deals mostly with hydrogen control measures and containment venting. Core melt-concrete interaction is mentioned there, but only in the context of hydrogen generation which can lead to combustible gas mixture. It is only in the design of NPPs built in the last decade that this issue has been addressed.

The recent WENRA pilot study on harmonization of nuclear safety in WENRA countries in the chapter on "Instrumentation and hardware provisions for SAM" includes the goal that "Means shall be provided to prevent containment melt-through", as one of the reference levels that reflect the highest quartile of existing national requirements [WENRA 03]. The study has indicated however, that "none of the WENRA countries totally complies with the reference levels", and adds that when "harmonisation measures are not judged to be reasonably practicable, an extended analysis of possible compensatory measures is recommended".

The resolution of the problem has not been yet reached due among other reasons to inherent difficulties in experimental simulation of high temperature chemical reactions of molten corium with internal heat sources in large-scale installations. In the past decades much experimental work was done to learn the relationships influencing MCCI processes. The aspects studied on various experimental stands were the composition and temperature of molten corium, composition of concrete and influence of rebars in the basemat, and possibilities of top cooling of corium with water. In the US these experiments were used to validate the code CORCON [ERI 93], which is presently used as the module for MCCI calculations in MELCOR, and in European Union the validation was done for WECHSL code, used as MCCI calculating module in ESCADRE code system [TACIS 02].

Among the open questions is the issue of whether covering the basemat with water will be enough to stop corium-concrete interaction and eventual concrete penetration, or not. The experiments performed so far suggest that after corium quenching with water a thin oxide crust develops which prevents further cooling of the remaining liquid phase of corium. However, taking into account large dimensions of the corium pool it is expected that the crust will break, improving cooling of the molten corium layers beneath.

The experiments performed within the ECOSTAR programme at FZR Karlsruhe showed that the development of an oxidic crust prevents effective cooling by top flooding [Steinwarz 01, EUR20281]. Also in a large-scale test COMETPC-H4 the melt formed a surface crust under the water and prevented fragmentation and cooling of the liquid melt underneath. Bottom cooling seems to be promising, however it is technically not feasible to be installed in existing NPPs and therefore not envisaged. Large-scale 2-D experiments are planned within EU research programmes, but their results have not been published yet.

Thus, according to the present day state-of-the-art, it cannot be demonstrated that the strategy of covering reactor cavity with water and corium spreading will be sufficient to prevent basemat penetration.

Current plant status

The main strategy of Temelín NPP to prevent basemat penetration consists in filling up the reactor cavity with water before the RPV failure and opening the cavity door to facilitate corium spreading over an area of about 100 m^2. This is in accordance with recommendations of EU TSO [EUR 20163] that the spreading area should be at least 0,02 m^2/MW of reactor thermal power, which in the case of Temelín would yield an area of 60 m^2.

Both Czech calculations and PN7 analysis showed that corium spreading would significantly slow down the rate of basemat penetration. In the case of LB LOCA the time to basemat failure – without taking account of the water layer – was evaluated in PN7 as 48 hours without spreading and 74 hours with spreading over 100 m^2. Similar values were obtained in Czech calculations [Pazdera 03].

The analysis of water sources available in the plant and of the possibilities of flooding the reactor cavity has been made and the corresponding curve showing the achievable level of water has been established. It has been shown that the level of water can maximally reach up to the bottom of the RPV, but not to cool down the molten corium inside the RPV. Czech specialists estimate that the presence of water at the moment of RPV break would significantly reduce the initial MCCI rate [Sykora 03]. However, no details of the calculations or of underlying relationships considering the influence of water and slowing down or stopping the MCCI have been provided.

The Czech side provided a document indicating the composition of basemat concrete after the Prague Workshop. It was stated that the base concrete is the same in the layer below the reactor cavity and outside, below the equipment room. It is characterized by the absence of carbon compounds, which significantly reduces the amount of gases produced due to MCCI. The rates of concrete penetration in vertical and horizontal direction are calculated by CORCON module, which is a part of the MELCOR code system, validated against several experiments [ERI 93]. The calculations of basemat penetration made initially by PN7 team were repeated for updated concrete composition after the Prague Workshop [Sart 03b] and provided results comparable to those quoted by Czech side for the concrete penetration without influence of overlaying water layer. However, no stopping of concrete penetration was found under any conditions.

The technical problems related to spreading of corium are to be resolved by the Temelín NPP within the preparation of SAMG implementation in the plant. In order to ensure opening of the cavity door, a remote control device will be installed and the door is to be opened early in the accident sequence, much before the moment of RPV failure. Two movable barriers will be installed in the neighbouring containment room to prevent molten corium flowing to the equipment hatch (which would fail the containment integrity) and to protect the area from missiles that can be thrown from the hermetic door after RPV break.

An analysis of molten corium progression in the instrumentation channels has been performed and it was shown that the corium would quickly freeze and would not be able to flow down the channels [Kujal 03]. In addition, special protection of the channels is planned. Thus the general features of the strategy to prevent basemat penetration are defined and look feasible.

In answer to questions, NPP Temelín experts acknowledged that there is a possibility of improving leak tightness of the rooms below the containment basemat and sealing them off in case of a severe accident involving basemat melt through. The provision of leak tightness of rooms adjacent to reactor containment is a recognized measure, similar to the measures recommended already in Rasmussen report [WASH 1400] where it was required that any possible leakages bypassing the containment (V-sequences) should be captured in closed rooms and not released directly to atmosphere. This strategy however, has not been studied in Temelín in detail yet. Temelín NPP stated that it prefers to concentrate on prevention, and this prevention includes:

- Prevention of DBAs from developing into BDBAs by following so EOPs,
- Prevention of RPV failure by RCS depressurization and water injection,
- Prevention of basemat penetration by covering the floor of reactor cavity with water and corium spreading.

The publications of Czech specialists include statements that the basemat integrity can be kept even after RPV break, for example in Ref. [Sykora 01], there is a statement that with delayed ECCS recovery the depth of concrete ablation after 300 000 seconds (~ 3,5 days) will be only 1,4 m. However, in view of the uncertainties connected with the top water-cooling of molten corium pools the issue cannot be judged closed.

The situation in Temelín is more critical than in a typical PWR NPP, because the basemat is situated about 13 m above the ground, and in case of its penetration the containment atmosphere will get in direct contact with the outside atmosphere. This can result in air mixing with the hydrogen accumulated inside the containment, which could not burn in view of inerted atmosphere, but will be able to burn or even detonate once it gets in contact with the free air, either outside or inside the containment. The analyses of the consequences of such a burn are not known so far, but should be a part of emergency planning.

Evaluation

While the measures already implemented and being planned by Temelín NPP go in the right direction, the PN7 team considers that they do not assure protection of the basemat if RPV failure occurs. The probability of RPV failure is small, as shown by recent PSA, but it exists. After RPV failure there are no visible means of stopping the basemat penetration completely, and the planned measures assure only slowing down of the process. Therefore, additional measures aimed at improving leak tightness of rooms below the containment basemat should be considered.

Inasmuch as the SAMG strategies include a capability for filtered venting, the potential for using filtered venting in connection with mitigating basemat failure could be considered. Use of filtered venting before basemat failure would reduce the containment pressure, so that once the basemat fails it would be less likely that the reactor building structure would fail due to overpressure. In addition, venting would reduce the amount of hydrogen in the containment atmosphere; also making it less likely that hydrogen combustion in the reactor building after basemat failure would result in reactor building structural failure. This approach could help preserve the reactor building as an independent fission product barrier (or at least as a function of hold up and attenuation of the source term before release to the environment).

Further monitoring should cover the implementation of technical means for:

- Timely opening the rector cavity door before the RPV break,
- Installation of removable shielding walls to restrict molten corium pool area and protect the containment liner and other barriers against molten corium and missiles hazards,
- Protection of containment penetrations and reactor cavity instrumentation channels against molten corium penetration.

The monitoring should also include the new set of analyses to be performed for Temelín NPP scheduled for the end of 2003, in particular the evaluation of possibilities to stop the MCCI, the estimates of possibilities of hydrogen burn in case of basemat failure and resulting revolatilization of fission products deposited inside the containment, and the effectiveness of possible reduction of leakages through the rooms under the containment basemat.

The hazards due to radioactive releases in case of basemat melt-through are much smaller than in the case of early containment break, as discussed in section 5.6 below. As shown in TACIS programme, the mass of radioactive aerosols still suspended in the containment atmosphere is dramatically smaller than the mass released from the core to the containment and available for release in case of an early containment failure. The radiological hazards are correspondingly reduced.

The strategy of Temelín NPP should therefore prevent re-volatilization of the fission products that have already been deposited on containment and piping surfaces, which could result in case of violent air turbulence. This includes prevention of massive hydrogen deflagration and detonation after basemat melt-through and mixing of hydrogen in the containment with the air from outside atmosphere.

Moreover, the pressure in the containment should be reduced before the basemat melt-through to avoid sudden air expulsion from the containment to the environment and carry-over of re-volatilized fission products. If the pressure in the containment is lowered in time, it will be also possible to minimize leakages through the rooms below the containment, between the basemat and the environment. During the Prague meeting Czech specialists mentioned these issues, but no detailed information was obtained on the approach to be followed.

The development of SAM strategy addressing specifically the issue of minimization of radiological releases in case of basemat melt-through should be monitored. Further discussion of this issue is provided in section 5.6.

5.4.3 Long term over pressurization of the containment

VLI No.	VLI title / description
8.7.1	What is the overpressure assumed to be the failure pressure of the containment? Is the strength of ventilation duct valves equal to that of the containment walls?
8.7.2	Is a filtered venting system available to be used in the case of loss of heat removal capability from the containment?
8.7.3	If there is a purge system, how is it protected against leaks outside the containment?
8.7.4	Do inherent passive features – like PARs – or by active technical equipment, provide containment mixing?
8.7.1a	What are the differences between the containment design at Temelín and that at typical WWER 1000 e.g. in Balakovo?
8.7.1a	CEZ-ETE's presentations in April 2001 and September 2001 address a station blackout sequence, and conclude that the maximum pressure in containment is 830 kPa (0,83 MPa) and report that containment integrity is "not challenged". The Temelín containment failure curve provided in the September 2001 viewgraphs however indicates a 5th percentile failure pressure of 0,8 MPa for the "wall-upper ring junction" failure mode. Given that the station blackout analysis shows a 0.83 MPa maximum pressure and there is a 5% chance of containment failure at about this pressure, what is the basis for concluding that containment integrity is "not challenged"?
8.7.2a	Is there a deliberate containment venting strategy (for pressure management, hydrogen concentration management, or other purposes) as part of the Temelín SAMGs, and if so, what is that strategy, what cues cause its implementation, and how does that strategy account for competing accident progression processes (i.e., how does the venting strategy consider reactor cavity melt-through to avoid melt-through under pressurised conditions)?
8.7.5	What is the physical and experimental basis for the conclusion that parallel hydrogen burning and DCH are not predicted for Temelín in case of HPME/DCH due to vessel failure at high pressure in a severe accident?

State-of-the-art requirements and practices

Large dry containments are recognized to have considerable safety margins, which provide effective protection against over pressurization both in short and long term. Nevertheless, in case of long-term failure of heat removal systems, the pressure inside the containment can grow to the values exceeding design strength limits. Therefore, several European countries have decided to install filtered venting systems, which can serve for controlled release of gases from the containment with effective retention of volatile fission products. Such filtered venting systems are installed e.g. in Sweden (FILTRA system), France (sand filters), and Germany (HEPA and charcoal filters in the venting system).

In the US the generic severe accident management guidelines developed by each NPP supplier owners group include either purging and venting, or only venting of the containment to address combustible gas control. On the basis of the industry wide commitment the NRC is not proposing to require such capabilities, but continues to view purging and /or controlled venting of all containment types to be an important combustible gas control strategy that should be considered in a plant's SAMGs. [NRC 02, NRC 03].

The analyses performed within TACIS programme for WWER 1000 NPPs showed that the maximum pressures achieved in the course of severe accidents reach from 0,425 MPa [Schoels 02] to 0,56 MPa [TACIS 02] for LB LOCA 850 mm and to 0,515 MPa for SB LOCA 80 [Schoels 02]. In case of virtual combustion of hydrogen the peak pressure in the containment can reach up to 1,3 MPa [TACIS 02].

Current plant status

The comparison of the containment cumulative failure probability functions for typical WWER 1000 [Morozov 03] and for Temelín NPP containment [POSAR] indicates that the Temelín containment has much better strength properties than the typical WWER 1000 containment considered in TACIS programme. The fifth percentile of Temelín containment cumulative failure probability (CFP) is exceeded at the pressure of 0,8 MPa (absolute), and the median CFP is reached at the pressure of 1,1 MPa (absolute). The strength of all containment elements including ventilation ducts is designed and tested to be the same as the strength of containment walls. Thus the Temelín containment can successfully withstand pressure increases that would be dangerous to other WWER 1000 NPPs. However, the calculated containment pressures in the case of blackout reach the value of 0,83 MPa, so they exceed the 5^{th} percentile failure pressure. CEZ stated that the containment integrity "is not challenged", but no explanation of this was provided during the meeting.

In the long term the main hazards are due to loss of heat removal and to gas generation from molten corium concrete interaction. As the concrete used in Temelín for the base layer of the containment basemat includes no carbon, there is no long term generation of CO and CO_2 and therefore the rate of gas pressure increase inside the containment is much slower than in other NPPs. Czech analyses include the case of CSS failure and evaluation of the rate of containment pressure increase due to residual heating of fission products. They show more than 24 hours are available before containment pressure reaches values dangerous for containment integrity. According to NPP Temelín, it is unthinkable that the containment spray system would not be recovered, at least in one train, during such a long time. Moreover, there is a possibility of containment venting, although this is regarded as the last measure to be applied [Sykora 03].

Temelín containment has no venting system designed specifically for severe accidents, and normal ventilation system has large pipes (400 mm diameter), which are not qualified for accident conditions. Therefore, this normal ventilation system would not be used in case of severe accidents. On the other hand, there is a system qualified for 1,6 MPa, provided with filters of high efficiency (99% for aerosols), with double lines and throttling valves, designed to be used for testing containment strength, but available in severe accident conditions. According to Temelín NPP, this system will be used as filtered venting system as a last resort, in case when other means of pressure control fail [Sykora 03]. No details concerning venting implementation threshold and its effectiveness were given in the Prague Workshop, nor are available in open literature, but the general features of the venting strategy indicate that it can be implemented and would resolve any problems with long term pressure increase inside the Temelín containment.

The containment mixing is provided by the containment spray system, and in the case of spray failure, by PARs, which generate considerable amount of heat and thus provide local natural circulation of the air. There is no other technical equipment that could be relied on to mix the containment atmosphere under accident conditions. The mixing of atmosphere in the SG boxes with the atmosphere in the containment dome is less efficient than in large containments of PWRs. This issue has been studied within PN7 project by means of 3-D GASFLOW code. The results will be discussed in section 5.5.

Evaluation

The measures listed above in conjunction with appropriate SAMG strategies are sufficient to consider the hazards of late confinement failure as being of negligible importance. The capacity of the filtration system, which is relied on in venting to reduce containment pressure, needs further attention in the context of further monitoring or in bilateral meetings between the Austrian and Czech governments.

5.5 Hydrogen Control

VLI No.	VLI title / description
8.8.1	What is the capacity and layout of PARs?
8.8.2	What is the status of PAR qualification?
8.8.3	Which poisons have been considered in PAR qualification?
8.8.4	Is there a regular PAR-testing programme to avoid their deterioration with time?
8.8.5	Have been any PAR tests done under realistic conditions of severe accidents?
8.8.6	Has the risk of PAR introducing a hydrogen burn been evaluated? What is the safe operating range? Has the consequences of hydrogen combustion ignited inside the catalytic recombiner been evaluated (missile effects)?
8.8.7	Has there been an evaluation of the form and temperature of the gas mixture at the PAR outlet performed from the standpoint of protection of safety related components?
8.8.8	Has the system of PAR management over the lifetime of the plant been developed?
8.8.9	Is the hydrogen concentration measured? How? What is the measurement range? Are the monitors environmentally qualified? Are the monitors classified as safety related components?
8.8.10	Is there a strategy developed for the late stage of the accident when the fraction of hydrogen is high and the steam provides inertization of the containment? How is the stable safe state to be achieved?
8.8.11	Has the danger of deflagration to detonation transition been considered for the ventilation ducts connecting reactor cavity with the containment?
8.8.12	Has the possibility of local hydrogen concentrations exceeding average values been considered? E.g. after the core cavity door failure and local mixing of hydrogen with air?
8.8.13	Have local hydrogen deflagrations in reactor cavity been considered?
8.8.14	What are the sensors and their measurement ranges installed in Temelín NPP for SA management purposes?

State-of-the-art requirements and practices

- *Requirements*

US NRC and some regulatory bodies in EU countries consider hydrogen hazards in NPPs with large dry containment to be negligible, but other EU countries regulatory bodies require technical means for hydrogen control.

After more than 20 years of studies of hydrogen hazards, NRC staff has recently published the results of analyses of hydrogen control issue for severe accidents that led the NRC to the conclusion that large dry containments have a significant capacity for withstanding pressure loads associated with combustion. In Individual Plant Examination (IPE) Program [NUREG 1560] and in [NUREG 1150] it was found that H_2 combustion is not a contributor to early failure for large dry containments. The conclusion was: it is not required to install a hydrogen control system in large dry containments. However, mixing was still considered necessary to prevent creation of local pockets of high hydrogen concentration.

According to the estimates quoted in Ref. [NRC 2000, p. 4-9], conditional large early release probability (CLERP) before the vessel breach is <<0,1 [1/a], CLERP at vessel breach is less than 0,1 [1/a] for high pressure path and <<0,1 [1/a] for low pressure path. After vessel breach CLERP is less than 0,1 [1/a]. The analyses made for Zion showed that CCFP before and at vessel breach was about 0,01 and the contribution to this low probability from hydrogen combustion was very small [NRC 2000, p. 4-10]. The results for Surry were similar to those for Zion. The NUREG-1150 study did develop uncertainty distributions and the 95th percentile for Surry was predicted to be 0,1 [1/a], and for Zion 0,05 [1/a]. The contribution of hydrogen combustion to these two estimates was again predicted to be small. This implies that even when the uncertainties are taken into account hydrogen combustion is not a major cause of containment failure before or at the time of vessel breach for large dry containments.

NRC has introduced rule changes that eliminate requirements of hydrogen control system in NPPs with large dry containments for design basis accidents. The Commission believes that accumulation of combustible gases beyond 24 hours can be managed by the licensee implementation of the severe accident management guidelines (SAMGs) and has therefore eliminated the requirements that necessitated the need for hydrogen recombiners and the backup hydrogen vent and purge systems [NRC 04].

NRC has also considered the request to eliminate the existing requirement in § 50.44(b)(2) 10CFR50 to ensure a mixed atmosphere in containment. However, NRC did not agree with this request and the final rule also retains the § 50.44(b)(2) 10CFR50 requirement that containments for all currently licensed nuclear power plants ensure a mixed atmosphere. [NRC 04]

German Reaktor-Sicherheitskommission (RSK) studied the hazard of hydrogen burning under severe accident conditions for 5 years, and in 1994 published its recommendations [RSK 94] developed in co-operation with the experts from USA, France and Japan. RSK assumed that in the case of a severe accident all safety systems could fail. If later on the core cooling is recovered, water would come in contact with zirconium and it should be assumed that all zirconium present in the core would be oxidized with corresponding production of hydrogen.

RSK stressed, that the steam would provide inert atmosphere in the containment, but after steam condensation the hydrogen burning could occur under conditions of high hydrogen concentration. Therefore, the Commission recommended installing in NPPs with steel shelled containments passive catalytic recombiners, which can work in low hydrogen concentrations even in the presence of steam, support atmosphere mixing and provide long term hydrogen removal. RSK observed that it is not realistic to expect that recombiners would be able to prevent in short term high concentrations of hydrogen and discussed possible application of dual hydrogen removal concept, with igniters and recombiners. Acknowledging the complexity of this area, the RSK formulated recommendations that for beyond design basis accidents the NPPs should install recombiners, and the capacity of the recombiners should be such that they would reduce the amount of produced hydrogen within several hours and thus contribute to reduction of containment failure risk. [RSK 94]. The operation of recombiners should be verified by tests conducted every year on randomly chosen catalytic elements.

For new NPPs the European TSO position states, that the containment volume and the mitigation means must be such as to prevent the possibility of global hydrogen detonation. The possibilities of local high hydrogen concentration must be prevented as far as achievable by the design of the internal structures of the containment, when it will not be possible to demonstrate that the hydrogen local concentration remains below 10%, specific provisions must be implemented such as inertisation or reinforced walls of corresponding compartments and of the containment. [Manuel 95]

In the long-term ex-vessel phase, autocatalytic recombiners should be able to handle the hydrogen generated by MCCI and water radiolysis in the sump area.

In order to eliminate the risks of detonation, the efficiency of the mitigation means has to be such that with a hydrogen production corresponding at least to 100% fuel clad metal water reaction coupled with an appropriate kinetics of release, the local concentration of hydrogen in the containment should be lower than 10% per volume. It must be verified in addition that a global deflagration of a total amount of hydrogen corresponding to these criteria does not endanger the containment integrity [EUR 20163].

- *Technical means of hydrogen control*

The main technical means to reduce the amount of H_2 in a closed containment are igniters and recombiners. Besides that, especially in large dry containments, which can withstand hydrogen deflagration without losing their integrity, natural convection driven atmosphere mixing helps limit the formation of pockets of high hydrogen concentration. When the hydrogen concentration is high, containment atmosphere can be kept inert by avoiding steam condensation. Finally, hydrogen can be removed from the containment by venting, which is implemented in several EU countries. In the case of large dry containments however, the hazard of hydrogen burning is generally considered to be small.

Available igniters exist in the form of spark igniters, catalytic (i.e. passive) igniters and sparkplugs. They are designed to ignite gas mixtures as soon as these mixtures have reached the flammability limit (e.g. 5% hydrogen and 95% air, or e.g. 20% hydrogen, 30% air and 50% steam). The corresponding pressure increase will lie in the order of some 0,1 Bar depending of course on the hydrogen concentration at the moment of ignition.

With the exception of catalytic igniters, spark igniters and sparkplugs have to be initiated. The existing spark igniters are passively initiated when either the surrounding temperature or the pressure increase beyond a certain limit. Active initiation by the operator, or actuation after measurement of hydrogen concentration shows that a certain limit is exceeded, can be also envisaged.

Catalytic recombiners exist in various forms and sizes. The volume of the system is about 0,5÷1 m^3; the flow area typically 0,5 m^2 while the steady state flow velocity attains around 1 m/s, independent of hydrogen concentration.

The removal rates depend on the recombiner size, number and location. In most EU NPPs the total capacity of recombiners, even those designed for severe accident conditions, is much lower than the maximum hydrogen production rates. For example, in German NPP Biblis B the total maximum capacity of all recombiners is 0,05 kg/s, and similar rates of recombination are ensured in other German NPPs. This is in keeping with the recommendations of RSK, which indicated that the capacity of recombiners should be sufficient to control hydrogen concentration in long term, not at the moment of the maximum rate of hydrogen production. In French NPPs the total capacity of recombiners planned to be installed in 1300 MWe PWRs is about 216 kg/h, which corresponds to 0,06 kg/s. There are, however, also such plants as Doel in Belgium with much higher recombination capabilities, and Spanish (except Trillo) or US plants where no recombiners are deemed to be necessary.

Atmosphere inerting is an inherent feature of many scenarios with loss of containment spray system. In some instances the containment atmosphere steam inerting has been demonstrated to provide a hazard self-limiting feature so, that the excessive hydrogen production resulting in elevated hydrogen concentrations in the containment can be outweighed by the steam produced from coolant.

Since this behaviour is rather sensitive to a number of boundary conditions, including uniform hydrogen distribution, it cannot be considered a perfect solution. It is by no means an inherent safety feature as suggested in several instances, because it does not return the plant intrinsically into a safe state avoiding incident or accident conditions like inherently safe installations would do. The need for producing steam to overcome excessive hydrogen production can paradoxically lead in some cases to shifting the critical safety function “cool the fuel” to a quite different safety function “avoid internal hazards” in which the cooling procedures are changed according to the inerting needs.

Severe accident conditions can result in hydrogen concentrations of a few percent up to 12-15%, but as long as there is plenty of steam in the containment atmosphere there is no danger of hydrogen detonation. In the process of steam condensation the volumetric fraction of hydrogen increases, but before it can come to detonable concentrations, deflagration is expected to occur [Blanchat 97]. In the TACIS calculations for WWER 1000 units, significant hydrogen concentrations reaching up to 12% vol. were found in several scenarios. Keeping containment atmosphere inert was one of the principal measures to deal with such situations. In order to maintain mixed atmosphere within the containment the main technical means consists in having containment fan units. This is an inherent feature of US hydrogen control strategy, which corresponds to the WOG-SAMGs being implemented at Temelín NPP.

Current plant status

The strategy of NPP Temelín in dealing with hydrogen hazards consists in prevention of detonable hydrogen concentrations by frequent hydrogen deflagrations and reduction of the hydrogen concentration by means of recombiners. There are 22 catalytic hydrogen recombiners (PARs) of Siemens make, Model No FR90/1-150, distributed within the containment with due consideration to the expected distributions of hydrogen in the containment atmosphere for design basis accidents. The capacity of these recombiners has been chosen so as to control hydrogen concentration after Design Basis Accidents, and in the case of severe accidents the recombiners provide long-term reduction of hydrogen concentration and depletion of oxygen, thus contributing to inertization of the atmosphere.

The producer has extensively studied the capacity of PARs under accident conditions. The relationships defining PAR capacity in relation to hydrogen volumetric concentration and atmosphere pressure are proprietary and have not been made available to the Austrian side by Temelín NPP. In preliminary calculations of PN7 project the threshold of PAR operation was taken as 0,5 vol.% concentration of H_2 and the PAR capacity was determined according to the available correlations. Recently, an update of PAR capacity curves was made in PN7 calculations, and the peak capacity of 1 PAR under LOFW or LB LOCA conditions was shown to be below 0,18 g/s for hydrogen and below 0,1 g/s for CO. For 22 available hydrogen recombiners at Temelín, the total nominal recombiner capacity is approximately 4 g/s. The recombiners in this case were designed for design basis accident (DBA) conditions. The range of hydrogen recombination values for other European PWRs with large dry containments is from zero (for plants without recombiners) to appximately 50 g/s for plants with recombiners designed for severe accident conditions.

The comparison of maximum rates of hydrogen releases during severe accidents with the maximum recombination rates provided by the existing PARs made within PN7 shows that the role of PARs in controlling short-term hydrogen concentration is small. The main impact of PARs is on long-term oxygen depletion and atmosphere inertization. However, their overall recombining capacity is small and thus the depletion is a very slow process. Under severe accident conditions, PARs of larger capacity would be desirable.

In Czech calculations made for Temelín for Medium Break LOCA with spray system in operation and without recombiners a large number of successive deflagrations (41 during 12 hours) were predicted [Kujal 03]. They resulted in maximum pressure peaks reaching 0,25 MPa, with mass of hydrogen burnt equal to 1150 kg. In the case with recombiners, deflagrations were also predicted to occur, but the mass of hydrogen burnt in deflagrations was less (630 kg) and the mass removed by recombination was 870 kg, so that after 12 hours the amount of H_2 remaining in the containment atmosphere was only 200 kg out of 1700 kg released into the containment.

The Czech assumption regarding hydrogen deflagrations is not convergent with the state-of-the-art, because there are neither igniters nor other technical means designed to initiate hydrogen burn. Temelín staff stated that it would be enough to actuate any mechanical devices with electrical drives to initiate hydrogen burning, and that the recombiners would initiate burning themselves. This is possible, but not certain. (Tests of the PARs installed at Temelín indicate that with sufficient oxygen and not too much steam, the PARs *always* initiate a deflagration once the hydrogen concentration in the PAR inlet reaches about 7%.)

Moreover, the distribution analyses, when performed with the nodalisation concept as presented, does not allow to base the effects of combustion on a purely numerical assumption on automatic ignition, in case the concentrations “homogenised” within the control volumes reach flammability conditions.

In Czech calculations the case of superposition of hydrogen deflagration together with DCH was considered and shown to result in the maximum containment pressure of 0,45 MPa, so below the design strength of containment [Kujal 03]. This conclusion is confirmed by the calculations performed within PN7 for LB LOCA, which showed that even with a large amount of hydrogen in the containment atmosphere (of about 800 kg), an unplanned deflagration would not increase the containment pressure above the design strength value.

Inertization of containment by steam is the main factor preventing hydrogen burns. It is a natural phenomenon, because the release of hydrogen always occurs together with the release of steam, so it can be regarded as an inherent safety feature of the plant. The calculations conducted within PN7 indicate rapid increases of steam fraction in the containment, so that the concentration of steam providing inert atmosphere is reached before high average concentrations of hydrogen are possible. Within some periods of time hydrogen can burn (concentration above 4,1%) but its concentrations are far below deflagration to detonation transition (DDT) level.

According to the position of European Union Technical Support Organizations (TSO) there is no hazard of dry large containment failure if the average concentration of hydrogen in its rooms is below 10 vol.%. According to Czech calculations, this condition is fulfilled with a large safety margin in Temelín NPP. However, this is true only as far as average concentrations in the rooms are concerned. The modelling of 3-D hydrogen distribution made with GASFLOW provides more information on local hydrogen distribution indicates that local gas clouds with higher hydrogen concentrations are possible resulting from trapping in the steam generator compartments in case of small LOCAs. The calculations made with GASFLOW and a model including 52 080 calculation nodes for Temelín NPP showed that for the hydrogen release rate of 0,2 kg/s there are some volumes within steam generator room where hydrogen reaches mildly burnable concentration except for a small transient region with a more sensitive mixture. This region, within the steam generator box just above the hydrogen release point, does not last long, nor grow very large, and it is not therefore felt to pose a significant hazard. The hydrogen release value of 0,2 kg/s that was used for the initial GASFLOW calculation was the initial MELCOR data provided as input for GASFLOW calculations. It agrees well with the evaluation of NEA experts who wrote in the report on “In vessel and ex-vessel hydrogen sources” that “*It is commonly agreed that the hydrogen source rate typically about 0,2 kg/s for a large size PWR of 1000 MWe is sufficiently accurate”* as long as the core geometry remains intact [NEA 2001-15].

Later severe accident progression calculations for the in-vessel phase of the SB LOCA accident performed with the SCDAP-RELAP mechanistic code package, performed after the GASFLOW calculations were completed, indicate a similar total quantity of hydrogen produced for the small LOCA sequence but in a shorter time frame. This gave rise to a higher release rate, exceeding 0,75 kg/s for periods of a few hundred seconds. A sensitivity calculation performed with GASFLOW for a synthetic hydrogen source of 0,8 kg/s injected into the steam generator box in a WWER 1000 configuration gave rise to a much larger sensitive cloud.

In the ex-vessel phase of the accident the rates of hydrogen release depend on accident scenario and are from 0,001 kg/s (TACIS programme) to 0,1 kg/s (Temelín, [Svab 03]), and from 0,07÷0,16 kg/s (PN7). NEA indicates that a significant part of hydrogen is produced during the early phase of MCCI, while Zr is being oxidized [NEA 2001-15]. This quantity is clearly dependent on the level of Zr oxidation during the in-vessel phase of the accident. For a typical PWR the overall amount of H_2 released into the containment by complete oxidation of all Zr in-vessel and ex-vessel is in the order of 1000 kg. After depletion of Zr and its follow-on products, long term H_2 release during MCCI is governed by Fe oxidation with typical release rates of 4 g/s, which would continue over several days. However, the issue of hydrogen production by zirconium oxidation has to be dealt with as a whole (in-vessel and ex-vessel phases) and the main area of uncertainty, which requires further analyses, is the oxidation during the in-vessel accident progression [NEA 2001-15].

The quasi–steady state reached after the accident is characterized by high fraction of steam in the containment (above 53%), and a very low fraction of oxygen (below 5%) so that the atmosphere is inert. Reduction of the amount of hydrogen and eventual achievement of stable steady state is possible by the use of filtered venting system [Sykora 03].

Evaluation

In Temelín several factors contribute to the hydrogen safety: presence of a large dry containment, early inerting of the containment by steam and long term inerting by decrease of hydrogen and oxygen content due to the action of the installed hydrogen recombination system. The results of Czech calculations presented during the Workshop showed that the containment integrity would be kept even in case of unplanned actuation of containment spray system at the moment when the contents of hydrogen are the highest, causing hydrogen deflagration [Kujal 03]. Similar results were obtained in PN7 calculations.

On the other hand the note in Ref. [SONS 01] indicates that in the case of hydrogen detonation due to operator’s errors the integrity of containment can be lost and the corresponding radioactive releases are evaluated. SONS pointed out that the operators are thoroughly trained and unlikely to make mistakes in severe accident conditions. The plant considers that following the SAMGs practically excludes the hazard of containment failure.

Hydrogen management strategies are addressed in the Temelín SAMGs. Several strategies aid in limiting the threat to containment integrity posed by hydrogen combustion, including limiting the containment pressure, reliance on PARs to recombine hydrogen over the longer term, use of containment venting to reduce containment pressure and deplete the hydrogen source in the containment, and maintenance of steam inerted conditions in the containment when possible to suppress the possibility of hydrogen combustion.

A limited potential for energetic hydrogen combustion has been identified by a state-of-the-art CFD code calculation for a SB LOCA sequence for the WWER 1000 containment configuration (specifically involving hydrogen "trapping" in the SG boxes). In the context of Temelín however, the frequency of the accident analyzed has a low frequency of occurrence (3×10^{-7} per year) and the occurrence of a containment-threatening detonation event even under the conditions identified is by no means certain.

However, during monitoring the team has not been able to identify the technical means by which the Temelín NPP could achieve “mixed atmosphere” as required by e.g. US-NRC regulations, therefore the question of assuring mixed atmosphere remains open.

5.6 Reduction of Radioactive Releases

VLI No.	VLI title / description
8.9.1	What are the measures to reduce radionuclide releases?
8.9.2	Is there an accident management procedure for the reduction of the volatile organic iodide in the containment atmosphere?
8.9.3	Is there a qualified system to keep pH of water basic in the containment?
8.9.4	Has the decomposition of hydrazine due to severe accident radiation fields been considered?
8.9.5	Following a leak or breach in the primary containment, or following a bypass failure of the primary containment (e.g., steam generator collector leak, steam generator tube rupture, interfacing systems LOCA), radioactivity may escape to other parts of plant buildings (e.g., the reactor building, the condenser hotwell, etc.). How do the Temelín SAMGs attempt to mitigate leakages to and from these other buildings – what strategies are followed?
8.9.6	If the basemat fails due to molten corium concrete interactions (MCCI), and the reactor building retains its structural integrity, what is the capability of the reactor building ventilation system to reduce the source term resulting from basemat failure? What is your evaluation of the likelihood of maintaining structural integrity of the reactor building following basemat failure (considering pressure release from the containment, release of combustible gases from the containment and continued evolution of gases due to MCCI in the reactor building)?

State-of-the-art requirements and practices

The magnitude of radiological releases after containment break depends on several factors:

1. The size of the break in containment
2. Whether or not the sprays are operating (enhanced aerosol deposition)
3. Whether or not the release path passes through a pool of water (aerosol scrubbing)
4. The time margin between the release from fuel and the release from containment.

Therefore, not all containment failures lead to large releases. SAM strategies and technical measures used for reduction of fission product release from the containment include the use of containment sprays and addition of chemicals to increase pH of water, from the steam generators – steam release of the defect SG to the condensers or to the feedwater tank and increasing the water level on the secondary side of SG to provide scrubbing of fission products before their release to the environment, and generally injection of water into any path of fission product release to increase scrubbing and fission product deposition on the internal surfaces.

There are measures addressing specifically the issue of fission product releases, especially of iodine and caesium. Since the volatile organic iodide is most difficult to retain, special strategy should be developed to reduce it in the containment atmosphere. High pH assures effective partitioning of iodine and keeps iodine in the CSS water, but keeping water in basic conditions requires special qualified system of chemistry control under severe accident conditions. WWER 1000 units have the advantage of effective containment spray system with chemical additions aimed to keep spray water basic, but some of these additions can decompose in high intensity radiation fields. Analyses published in the past, e.g. for Dukovany NPP, addressed this issue and showed that it can be resolved.

Current plant status

SAM strategies in Temelín NPP include measures aimed at prevention of PRISE accidents and at reduction of radiological releases if PRISE accidents do occur, as discussed above in Section 5.1.

For all accident sequences with fission product releases to the containment atmosphere there is a containment spray system of high reliability. There is a qualified safety grade system to keep pH of water high. Additional sources of water are available to back up CSS operation. The Czech specialists claim that the late failure of containment does not result in high radiological releases because the operation of spray system will have removed fission products from the containment atmosphere before the containment (or basemat) failure occurs. This is in agreement with the findings of analyses performed for other NPPs and with the results of experimental studies performed within the EU [Morozov 03, Schoels 02].

The strategy of flooding the reactor cavity before RPV break assures that there would be layer of water above the molten corium, so that the fission products being released during MCCI would be retained in water in the process of gas purging. Although water may not be effective in stopping MCCI, it is certainly an effective means of reduction radiological hazards.

Containment venting is regarded as the ultimate means of containment protection, but it is unlikely to be used to prevent containment failure in view of the analyses showing that the pressure inside the containment would be below the containment design capacity until basemat failure occurs. If venting were really necessary, it will be done by means of a venting system provided with filters so that the releases of fission products will be appropriately reduced.

The results of calculations regarding the reduction of fission product inventory available for release in case of late basemat melt-through indicate high effectiveness of sprays and internal deposition of volatile fission products, but no details have been disclosed concerning possible fission product revolatilization due to hydrogen burning after sudden changes in containment atmosphere composition in case of basemat melt-through. Temelín NPP experts stated during the Prague workshop that additional measures could be implemented in the case of basemat penetration to reduce fission product releases to the environment through the rooms below the containment. No detail of such measures was given.

In the case of basemat failure due to MCCI, which could occur in the late phase of the accident, 3 or more days after the RPV rupture, the fission products would be deposited on inner surfaces of the containment. TACIS results indicate that if the spray system were available, the reduction of source terms would be by three orders of magnitude [Schoels 02].

Similar results are given in Czech studies [SONS 01], [Pazdera 03]. However, there is no analysis of possible effects of hydrogen burning or explosion, which is a potential threat in case of sudden mixing of outside air with the containment atmosphere after basemat melt-through. In case of an explosion the pressure wave could result in re-volatilization of a large part of particulates that have been deposited on containment surfaces. Although such a possibility is remote, it should be analyzed and the means for reducing possible radiological consequences of such a sequence should be considered.

Evaluation

The measures and strategies to reduce fission product releases are in keeping with the international practice. The only open issues are the reduction of radiological releases in the case of basemat penetration by molten corium as well as the potential for hydrogen combustion in the reactor building after basemat penetration. The Specialist's Team would recommend the Austrian Governement to consider monitoring of both the details of calculations and the means to reduce fission product releases in case of basemat melt-through.

6 CONCLUSIONS

The PN7 project aimed to clarify whether the measures already implemented and planned in Temelín address plant vulnerabilities in the severe accident area and whether they will provide a safety level similar to NPPs presently in operation in the EU and US. The report presents a clear picture of the Temelín NPPs behaviour in case of severe accidents, determines open issues and formulates proposals for further monitoring.

6.1 Overall Conclusions

6.1.1 Regulatory approach and practice

SUJB has required the plant to prepare and accomplish a program to deal with BDBAs, including estimation of plant vulnerabilities, proposed accident management procedures and the schedule of their implementation. The targets set for severe core damage frequency and for large off-site releases are to underscore 10^{-4} and 10^{-5} per reactor year, respectively, which is consistent with the INSAG targets for existing NPPs.

The development of SAMGs is performed by the utility. The regulatory body defines acceptance criteria and provides guidance to Temelín NPP, leaving enough flexibility for potential candidate actions to address specific challenges.

6.1.2 Temelín programme of severe accident management

The development and implementation of Temelín SAM programme has not been finalized, however, the whole process is well advanced.

The overall concept and approach to development/implementation of SAMG package was found to reflect the current good practice in the SAM area. The selection of plant specific SAM strategies has been based on the well-established generic approach developed by Westinghouse Owners Group. These generic strategies have been adapted to Temelín plant conditions based on a systematic process that reflects the current state-of-the-art in this area.

The programme is supported by severe accident analysis and plant specific PSA. However, there were some instances when the existing results of SA analysis were not properly incorporated into the PSA. It should be noted that also some SAM strategies, apparently the most recent, are not well supported by SA analysis. The interface between the PSA team and thermal hydraulic analysis team needs improvement.

The calculation tools used for SA analysis are similar to those used worldwide for the purpose of SAM and the team that has been responsible for calculations is competent. The existing analyses provide a reasonable basis for understanding plant specific vulnerabilities to severe accidents and the identification of AM strategies. Some of the existing analyses are old and do not necessarily reflect the current plant status and state-of-the-art in the area of SA codes, modelling and simulation. The plant is planning to improve these analyses using more current codes.

The PSA study for Temelín NPP includes Level 1 and 2. An IAEA mission has reviewed the first version of PSA and the resulting recommendations are reported be incorporated into the upgraded study. However, the upgraded PSA is still not finalized. Generally, the PSA study was developed in compliance with the current state-of-the-art. PN7 team has observed some deficiencies, but they are not expected to have significant impact on the final conclusions with regard to SAM strategies. The existing results have been used in the development of SAMG strategies and setting up priorities in the execution of strategies.

Westinghouse has developed a plant specific SAMG package in close co-operation with plant staff. The contents, structure, and format of plant specific SAMG, which were shown at the Prague Workshop, have been found to reflect the current state-of-the-art practice. This package is currently under internal review and translation into Czech language.

Organizational arrangements related to SAMG have not been finalized yet. Although the upgraded ERP Emergency Response Plan has been submitted to SUJB for approval, the updated version of the Emergency Operating Procedures including transition points to SAMGs need to be developed and implemented. Some concerns can be raised in the definition of responsibilities/authorities for determination and approval of an intentional release of radioactive material during a SA. The staffing of SAMG Evaluation Group within the Technical Support Centre is another issue that is not clear enough. The monitoring process by the Austrian Government should cover these aspects.

The plant properly considers all further steps of SAMG implementation including validation and training and plans for their execution are being developed. Based on the available knowledge all the related plant arrangements are considered adequate. Little is known also about the training and refreshing courses of SAM staff and the related schedules for implementation. However, the related activities should be subjected to monitoring by Austrian Government.

It should be noted that proper evaluation of the SAMG package including the supporting analyses would require detailed investigations that involve specialized expertise and considerable effort. Such evaluation was beyond the scope of PN7 project. Therefore, it would be very desirable to have detailed aspects of SAM development and implementation addressed by qualified independent external reviewers (e.g. IAEA RAMP mission). It is known that the plant management and SUJB seriously consider having independent review of SAM.

6.1.3 Technical measures available in Temelín for SA management.

One of the main areas of hazards due to severe accidents is that of primary to secondary circuit leakages, since such leakages involve loss of coolant accidents with the leak point situated outside the containment. In case of such accident all four barriers preventing radioactivity release to the environment can be lost simultaneously. Both contemporary regulatory guidance and industrial practice stress the necessity to avoid large PRISE events. In Temelín the hazards involved in primary to secondary leakage (PRISE) accidents are well recognized, the appropriate strategies developed, and the technical means provided to cope with PRISE events.

Another potential hazard is connected with long term complete loss of electric power, both from outside sources and from emergency diesel generators installed at the NPP. In such a case the means of heat removal from the reactor are lost, except for gradual evaporation of water, first in the secondary, then in the primary coolant circuit. If this situation persists for several hours the coolant in the core will evaporate, which leads to the core dry out, and damage.

The preventive measures at Temelín NPP correctly address the issue of station blackout. The most important measure for mitigation of the effects of blackout and other transients involving loss of electric power consists in forced depressurization of the primary circuit. The calculations showed that the capacity for depressurization in Temelín is well comparable with that in other plants of similar vintage. Moreover, the measures taken to prevent a blackout seem to be satisfactory.

The measures available in the plant are sufficient for timely depressurization of RCS. The Temelín NPP has two lines of defence in this respect (PORV and EGRS), which is better than in many other NPPs of similar vintage. The WOG SAM strategies being implemented in the plant recognize the importance of depressurization. However, while the capacity of PORV is fully sufficient for plant depressurization, the efficiency of the EGRS is just at the limit. If PORV should fail, the question of exact evaluation of EGRS efficiency would be important.

In view of the long delays of core damage in case of blackout, the limited capacity of batteries in Temelín seems to be inappropriate. According to the design the period of time that the batteries are sufficient for plant control is shorter than the time that would pass before severe damage of the core. Thus the potential advantages of good thermal hydraulic properties of Temelín could not be used due to battery limitations. Temelín EOPs and SAM strategies include measures to extend battery power supply time by re-structuring the load profile much beyond the design period of 1 hour. Nevertheless, it would be desirable to exchange batteries or include into the system additional power sources providing electric power during station blackout.

An important safety advantage of Temelín NPP is the fact that it is provided with a large dry containment. This reduces considerably the challenges to containment integrity during severe accidents. Similarly as in other NPPs with large dry containment, the hazards of early containment failure due to DCH in Temelín NPP have been evaluated as negligible and the strategy of RCS depressurization included in SAM in Temelín further reduces such hazards. The long-term pressurization hazards are reduced by the fact that the basemat concrete in place in Temelín does practically not contain any carbon, so there is no build-up of carbon monoxide and carbon dioxide due to molten corium-concrete interaction. This reduces the long-term quantities of non-condensable gases inside the containment. The calculations with the MELCOR code showed that the containment integrity is not threatened by long term increases of pressure due to gas generation and the presence of the containment spray system.

The monitoring orients itself on approaches used predominantly in some Western European countries. In the case of WOG SAMGs the basic work accomplished orients itself on the US-NRC position as introduced at the Temelín NPP.

During the monitoring according to the Melk Process no clarification could be found which rules and regulations were applied to consistently address the severe accidents related hydrogen issue at Temelín NPP.

Hydrogen hazards in NPPs with large dry containment are considered to be negligible by US NRC and some regulatory bodies in EU countries, but many EU regulatory bodies require technical means for hydrogen depletion. In Temelín the release rates of hydrogen during the in-vessel phase of the accident are comparable with those in PWRs, and the volume of the containment is similar. The geometry of the steam generator boxes and the ducts there is different from that in PWRs and makes hydrogen mixing less effective, which in case of SB LOCA in this area can lead to local formation of sensitive clouds of hydrogen during the in-vessel accident phase.

The likelihood of such an event is very low for Temelín (3×10^{-7} [1/a] for Temelín for small LOCA sequences; 1×10^{-7} [1/a] for medium LOCA; and 3×10^{-8} [1/a] for large LOCA), in total about 3% of the core damage frequency. In the ex-vessel phase the presence of a large dry containment and early inerting of the containment by steam contribute to prevention of hydrogen hazards. In the long term the installed hydrogen recombination system designed for DBA conditions, but passively operating also under severe conditions, will contribute to containment inerting by reducing the hydrogen and oxygen content. However, this process is slow and for severe accidents it would be advantageous to have properly located PARs of higher capacity.

In any case, the distribution analyses of hydrogen and steam inside the containment system should be performed with a more useful modelling concept taking into account also the operation of the spray system to evaluate more realistically the periods where inerting by steam may exist. Combustion should not be coupled to the distribution analysis without good reason (e.g., the activation of a reliable ignition source when small hydrogen concentrations are present).

The Czech strategy consists of early hydrogen deflagration, which should help prevent formation of sensitive clouds during in-vessel phase, and long-term inerting of containment with steam during the ex-vessel phase. Both Czech and PN7 calculations showed that in the case of unplanned actuation of the containment spray system at the moment when the contents of hydrogen is the highest the containment integrity could be lost, and Czech materials provide an evaluation of radiological consequences of such a scenario. However, the SAM strategy proposed for Temelín correctly addresses the issue of reduction of the hazards of late confinement failure due to hydrogen deflagration. In the case of ultimate necessity, Temelín can actuate the containment filtered venting system (normally used during pressure tests) to reduce containment pressure and hydrogen content. This issue seems to be still under development. As the heating due to fission product collection in filters can result in rising filter temperatures (with loss of filter efficiency) or in the worst case induce filter burning, the issues of filtered venting in Temelín should be further monitored.

The main hazard consists in the possibility of containment basemat penetration.

The measures planned be implemented in Temelín in case of RPV failure assure slowing down of the molten corium concrete interaction (MCCI) process. While these measures go in the right direction, it cannot be proved that they assure protection of the basemat against penetration by molten corium if RPV failure occurs. The likelihood of RPV failure is small, as shown by recent analysis, but it exists. According to the statements of Czech specialists, the measures planned in Temelín include corium spreading and water-cooling, which together should enable to stop the corium progression.

The calculations performed within PN7 project confirmed that corium spreading slows down the process and provides additional time margins. The effectiveness of water-cooling was not studied in PN7 due to the lack of access to the latest experimental OECD data. Recent information about the results of large scale tests on concrete penetration by molten corium conducted within OECD programme on “The Melt Coolability and Concrete Interaction“ indicates that in large scale test in the US enhanced cooling was obtained due to long term water cooling of the molten corium mass. The Czech Republic participates actively in this programme and has the actual information available.

As of now, the stopping of the corium ablation progress cannot be clearly demonstrated. Therefore, the Temelín staff considers additional measures aimed at improving leak tightness of rooms below the containment basemat. The hazards due to radioactive releases in case of basemat melt-through are much smaller than in the case of an early containment rupture. As shown in the TACIS programme, the mass of radioactive aerosols still suspended in the containment atmosphere at the time of basemat melt through is dramatically smaller than the mass released from the core to the containment and available for release in case of an early containment failure. Not considering re-volatilisation and emanation of deposited contaminants during late containment failure, the radiological hazards are almost correspondingly reduced.

The strategy under consideration of Temelín NPP includes prevention of re-volatilization of the fission products that have already been deposited on containment and piping surfaces, which could result in case of violent air turbulence. This includes prevention of massive hydrogen deflagration and detonation after basemat melt-through and mixing of hydrogen in the containment with the air from outside atmosphere. Moreover, it is planned to reduce the pressure in the containment before the basemat melt-through to avoid sudden air expulsion from the containment to the environment and carry-over of re-volatilized fission products. If the pressure in the containment is lowered in time, it will also be possible to minimize leakages through the rooms below the containment, between the basemat and the environment. During the Prague meeting Czech specialists mentioned these issues, but no detailed information was obtained on the approach being followed.

The measures and strategies to reduce fission product releases are in keeping with the international practice. The open issues are mostly connected with the reduction of radiological releases in the case of basemat penetration by molten corium. Czech specialists consider it a problem for future consideration, while they see as the most urgent tasks those, which are related to prevention of the basemat melt-through.

6.2 Recommendations for Further Monitoring

The monitoring process conducted so far within the framework of the Temelín Roadmap in the area of severe accidents helped to clarify a number of relevant issues. It was demonstrated that a comprehensive process directed towards accomplishing the comprehensive SAM and mitigation of SA consequences is in place at Temelín NPP. However, this process is ongoing and the Specialist's Team at the present can only follow a number of views and expectations on the SAMs final implementation as expressed by the Czech side.

Based on the recognition that the pertinent Czech-Austrian Bilateral Agreement is the appropriate framework giving the opportunity for further discussion and sharing additional information on these issues, the Specialist's Team would appreciate if the major findings could be revisited in the further monitoring process of the SAM. The following areas were identified as those, where additional information would be most valuable to consolidate the Monitoring results:

- The supporting severe accident analysis and PSA as well as their use in the verification of SAM strategies and the related procedures,
- SAMG implementation activities including procedural framework, SAMG validation, and SAM related staff training,
- Identification of the permissible degree of non-uniformities in the hydrogen distribution in the atmosphere,
- Implementation of plant changes to enhance the technical measures for SAM.

More detailed discussion of the proposed monitoring issues in these areas is provided below.

Calculations to be made by the operator/regulator are recommended on the technical level to be monitored jointly in the framework of the pertinent bilateral Agreement between Austria and the Czech Republic:

The calculations will be performed using the MELCORE 1.8.5 Version and are supposed to provide results on the following topics:

- Confirmation of capability of PORV to reliably depressurize the RCS (if its reliability at low pressure is lower than that announced during the Prague meeting, the credit taken for EGRS operation should be checked).
- Effectiveness of PAR recombiners under various severe accident conditions and related final states to be reached with SAM strategies planned in Temelín.
- Additional use of filtered venting for mitigation of radioactive releases.
- Filtered venting system operation, in particular the resistance of filters to heating due to the accumulation of fission products in those filters.
- Completeness of the analyses of corium attack on the basemat (the calculations should be carried out up to the moment of final basemat failure and include radiological consequences of this scenario).

SAMG implementation activities at Temelín should be further monitored to confirm that the remaining steps of the implementation process are successfully completed. Important items that need further monitoring/verification include:

- Confirmation that the revised procedural framework is implemented (E-plan, Emergency Implementation Procedures, and Emergency Operating Procedures).
- Confirmation that the responsibilities and authorities for intentional release of radioactive effluents in emergency conditions (as one of the SAM strategies) are clearly defined and understood within the overall ER organisation.
- Extent and the results of the SAMG validation process including observations on the sufficiency of staffing and organisational structure of the TSC (SAMG Evaluation Group) and the related feedback.
- Extent / scope of staff training process in the area of SAM and the related feedback.
- The recommendations from any independent review of SAM and their resolution, if made available by the plant.

Technical measures needed for prevention and mitigation of risk significant scenarios should be monitored to demonstrate that appropriate plant arrangements are in place (both procedures and hardware measures). Due attention should be given to SA situations that are most relevant from safety point of view such as basemat penetration in case of molten corium release from the RPV and station blackout. The following aspects are worth to be mentioned in this context:

- Timely opening the rector cavity door before the RPV failure,
- Installation of removable shielding walls to restrict molten corium pool area and protect the containment liner and other barriers against molten corium and missiles hazards,
- Protection of containment penetrations and reactor cavity instrumentation channels against molten corium penetration.
- Increasing capacity of batteries so that they would be available throughout the blackout accident at least until RPV failure
- Assuring possibility of fire truck water deliveries in case of total blackout on the site.
- Evaluation of possibilities to stop the MCCI,
- Estimates of possibilities of hydrogen burn in case of basemat failure and of resulting revolatilization of fission products deposited inside the containment
- Effectiveness of possible reduction of leakages through the rooms under the containment basemat.

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8 ABBREVIATIONS

AC	Alternative Current
ADP	Administrative Procedure
AMP	Accident Management Programme
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design Basis Accident
BO	Blackout
BRU-A	Steam Dump Station/Valve to the Atmosphere
CA	Computational Aids
CDF	Core Damage Frequency
CET	Core Exit Temperature
ČEZ	Česke Energeticke Zavody (the Czech Electricity Generating Company)
CLERP	Conditional Early Release Probability
CLLRP	Conditional Late Release Probability
CONT	Containment
CSS	Containment Spray System
DBA	Design Basis Accident
DC	Direct Current
DCH	Direct Containment Heating
DDT	Deflagration to Detonation Transition
D_{eff}	Effective Diameter (flow area related)
DFC	Diagnostic Flow Chart
DSMG	Decision Making Support Group
EC	European Community
ECCS	Emergency Core Cooling System
ED	Equivalent Diameter (break dimension size)
EFWS	Emergency Feed Water System
EGR(S)	Emergency Gas Removal (System)
EIP	Emergency Implementation Procedure
EOF	Emergency Operating Facilities
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ERP	Emergency Response Plan
ETE	Nuklearna Elektrarna Temelín (Czech abbreviation)
EU	European Union
FP	Fission Products
FVS	Filtered Venting System
HP	High Pressure

HPI(S)	High Pressure Injection (System)
HPME	High Pressure Melt Ejection
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IDCOR	Industry Degraded Core Rulemaking
IE	Initiating Event
INPO	Institute of Nuclear Power Operations
INSAG	International Nuclear Safety Advisory Group
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IPSART	International PSA Review Team (IAEA Mission, former IPERS)
IRRT	International Regulatory Review Team
IS LOCA	Interfacing Systems LOCA
LB LOCA	Large Break LOCA
LBB	Leak Before Break
LER(F)	Large Early Release (Frequency)
LOCA	Loss of Coolant Accident
LOHS	Loss of Heat Sink
LOSP	Loss of Station Power
LP	Low Pressure
LPI(S)	Low Pressure Injection (System)
LRF	Large Release Frequency
MAAP	Modular Accident Analysis Program
MCCI	Molten Corium-Concrete Interaction
MCR	Main Control Room
NDT	Non Destructive Test
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant
NUSS	Nuclear Safety Standards
OECD	Organisation for Economic Co-operation & Development
OSC	Operation Support Centre
PAR	Passive Autocatalytic Recombiner
PDS	Plant Damage States
PIE	Postulated Initiating Event
PM	Project Milestone
PMR	Preliminary Monitoring Report
PORV	Power Operated Relief Valve
POSAR	Preliminary Operating Safety Analysis Report

PRISE	Primary to Secondary Leakage
PRZ	Pressurizer
PSA	Probabilistic Safety Assessment
PTS	Pressurized Thermal Shock
QA	Quality Assurance
RAMP	Review of Accident Management Programmes (IAEA Mission)
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RSK	Reaktor-Sicherheitskommission (Germany)
RWST	Raw Water Storage Tank
SA	Severe Accident
SACRG	Severe Accident Control Room Guidelines
SAG	Severe Accident Guidelines
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SAMP	Severe Accident Management Programme
SAR	Safety Analysis Report
SB LOCA	Small Break LOCA
SBO	Station Blackout
SCG	Severe Challenge Guidelines
SCST	Severe Challenge Status Tree
SG	Steam Generator
SGCL	Steam Generator Collector Leakage
SGTR	Steam Generator Tube Rupture
SIR	Special Information Request
SIT	Safety Injection Tank
SO EOP	Symptom Oriented EOP
SONS	Regulatory Authority of Czech Republic (English abbreviation)
SSC	Systems, Structures, and Components
SUJB	Regulatory Authority of Czech Republic (Czech abbreviation)
TK	Make up system (plant specific system identifier)
TSC	Technical Support Centre
TSO	Consortium of European Technical Support Organizations
TSO	Technical Support Organisation
ÚJV	Nuclear Research Institute (Czech abbreviation)
US NRC	US Nuclear Regulatory Commission
V&V	Verification and Validation

VLI	Verifiable Line Item
WANO	World Organization of Nuclear Operators
WEC	Westinghouse Electric Corporation
WESE	Westinghouse Energy Systems Europe S.A.
WOG	Westinghouse Owners Group
WWER	Russian design of PWR

9 UNITS

°C	Celsius (degrees) (temperature): 0 [°C] = 273,6 [K]
°C/h	Celsius per hour (temperature change with time) 1 [K/h] ≡ 1 [°C/h]
1/a	Events per year (frequency of events per year (of reactor operation))
Bar	Bar (pressure difference) 1 [Bar] = 10E5 [Pa] (in excess of environmental)
Bar_{abs}	Bar absolute (pressure absolute) 1 [Bar] = 10E5 [Pa]
cm	Centimeter (length)
cm/h	Centimetres per hour (speed (here speed of ablation))
d	Day (time)
g	Gram (weight)
K	Kelvin (degrees) temperature or temperature difference
K/h	Kelvin per hour (temperature change with time) 1 [°C/h] ≡ 1 [K/h]
kg/s	Kilogram per second (mass-flow)
kJ/mol	Kilo-Joule/Mol (work per Mol - chemical reaction work)
km	Kilometer (distance, length)
kPa	Kilo-Pascal
m	Meter (length)
m^2/MW_{th}	Specific surface for heat transfer
m^3	Cubic meter (volume)
mm	Millimeter (length)
MPa	Mega-Pascal
MPa_{abs}	Mega-Pascal absolute (pressure)
MW_e	Electrical power output/demand
Pa	Pascal (pressure) 1 [Pa] = 1[N/m²]
s	Second (time)
Sv	Sievert (effective dose (received by humans from radioactive radiation))
Sv/a	Sievert per year (of operation) Risk to the public resulting for one year of operation
t	Ton (weight)
t/h	Tons per hour (mass flow)
vol.%	Fraction of volume (gas)
wt.%	Fraction of weight (solids, liquids) or also denoted as %
y	Year (time)

ANNEX A

SAM DEVELOPMENT AND IMPLEMENTATION IN WESTERN EUROPE AND THE USA – POSITION – OUTLOOK

In order to collect in a brief overview the rather heterogeneous AM approaches embarked on by the various countries operating NPPs we summarise here the main aspects of the state-of-the-art in severe accident management at NPPs in Western Europe. This should enable also to conclude about the European position of the SAM approach at Temelín NPP.

For this purpose the statements associated with the individual monitoring topics above will be taken into consideration in the first place and then the overall position versus the European environment will be judged as it is described here in brief.

Many of the SAM practices in Western Europe are based on the earlier developments in the United States, where the four groups of owners of nuclear power plants each developed and implemented their proprietary SAMG approach. Inasmuch as Temelín has selected the Westinghouse Owner's Group (WOG) SAMG methodology, the discussion that follows will focus on that aspect. Comments on other approaches will be made as necessary.

History of SAMG development

In 1985, the USNRC issued a "Severe Accident Policy Statement" in which, based on then-available information, it judged that the existing NPPs were acceptably safe in their then-current layout and configurations [NRC 85]. Nevertheless, the NRC set in motion a four-pronged "integration plan for closure of severe accident issues" that identified areas where action by nuclear utilities was required [NRC 88 a]:

- Individual Plant Examination (IPE) for severe accident vulnerabilities
- Containment performance improvements (CPI)
- IPE of External Events (IPEEE) for severe accident vulnerabilities
- Accident Management (AM)

The IPE and IPEEE programs were initiated in 1988 and 1991 [NRC 88 b, NRC 91]. All NPPs were required to perform IPE evaluations (for internal events) and IPEEE evaluations (for external events). (Level 1 PSA largely met the IPE & IPEEE program requirements with a containment performance evaluation or Level 2 PSA; some events, especially seismic, were analyzed by a margins approach without quantifying the core damage frequency contribution.) The CPI program was defined in 1990 [NRC 90 b]. For large dry containment PWRs, it was concluded that utilities with such plants should evaluate containment and equipment vulnerabilities to localized hydrogen combustion and the potential need for improvements (including accident management procedures) as part of the IPE program.

Concerning AM, after the NRC informed the utilities that it expected a certain level of severe accident management at their plants, the nuclear industry committed itself in November 1994 to SAM programmes consistent with industry guidance [NEI 94]. The NRC accepted the industry initiative on accident management on 9 January 1995 (via a letter to the Nuclear Energy Institute). Implementation of SAM programmes in the US was completed between June 1997 and December 1998. The US SAM guidance approach established by the industry and accepted by the NRC was focused on the establishment of SAM guidance that made maximum use of existing plant capabilities, and with only minor modifications. This was consistent with NRC's earlier finding in the Severe Accident Policy Statement in 1985 concerning the acceptability of the NPPs in their then-current configuration and layout [NRC 85].

Implementation of US SAMGs

The SAM programmes adopted for US NPPs are characterized by very detailed guidance for all phases for a severe accident, up to and including RPV failure, basemat attack, and a variety of other potential challenges to containment integrity. The US SAM guidance is mechanistically based - that is, guidance was developed for all situations and phenomena that can physically arise, irrespective of their frequency of occurrence. The approach is based on the Electric Power Research Institute (EPRI) "Technical Basis Report" (TBR, proprietary), which encompasses the physics and phenomenology of severe accidents. The TBR, which analysed plant vulnerabilities and described potential countermeasures, which are available, analysed their potential effects during a variety of plant damage states, and incorporated research findings from the USNRC and industry as well as IPE and PSA insights. The largest SAM programme in the US was that of the Westinghouse Owners Group (WOG). The WOG SAMG approach was later selected by many European utilities, for both Westinghouse and some non-Westinghouse plants.

Plant modifications and SAMG adoption in Europe

In Europe, the picture of SAM and SAMG is quite diverse compared with the US situation. In response to the severe accidents at Three Mile Island in 1979 and Chernobyl in 1986, various European regulators had required or requested measures against severe accidents, with the focus on hardware measures. NPPs in a number of countries were required to install filtered vents to their containments. (Some sort of filtered venting capability was ultimately installed in approximately four out of five PWRs in Western Europe. These systems range from sand filters installed on the top of the auxiliary building to a variety of different filtration systems housed in separate, dedicated structures. Belgian, Spanish, and UK regulatory authorities have not required implementation of filtered venting systems.)

In addition, many NPPs in Western Europe installed passive autocatalytic recombiners (PARs), sized and designed for severe accident conditions, to prevent or mitigate hydrogen combustion/deflagration/detonation. Belgium was the first Western European country to require PARs, which were implemented in its PWR NPPs between 1995 and 1997. The current state-of-the-art in Western Europe for severe accident management includes severe accident-sized and designed PAR systems. Most PWRs in Western Europe have either implemented such systems, or have decided to do so in the past few years and are in the process of implementation [Bachellerie 03].

It should be noted that many plants in Europe have also built additional preventive features, such as additional redundant power supplies from adjacent stations, well-qualified PORVs for bleed and feed operation, bunkered systems for sustained heat removal under extreme external events (earthquake, air plane crash, extreme winds, acts of terrorism), feeding SGs from mobile source outside the plant, etc. This was done notably in Belgium, Germany, Netherlands and Switzerland. Some built also control room air filtering (Germany, Trillo in Spain). Loviisa built a separate primary depressurisation system (apart from the existing PORVs).

A number of plants have provided for cross-connections between systems within a unit (allowing, for example, containment spray pumps to provide low pressure injection or vice versa) or between systems in adjacent units (allowing, for example, the high pressure injection system in one unit to supply high pressure injection in the adjacent unit if that unit's system failed). Such modifications were often regulatory-driven, but were also done at the initiative of the utility for a variety of reasons (including investment protection). (The USNRC investigated the European efforts for application in the US, notably the bunkered systems, but concluded that such systems were not cost-beneficial; cost-benefit considerations are part of US "back fitting" legal requirements.)

European AM Procedures and Guidelines

The Procedures and Guidelines developed in Europe relate to the origin of the plants: Most plants of US-type in Europe were aware of the developments in the USA, and have implemented similar guidance, albeit it somewhat later than the US schedule. In many cases, this was promoted or even required by the regulatory body in these countries.

Non-US plants had initiated similar initiatives, but mostly on a different basis. This will be the subject of subsequent sections, where the situation in the different countries is further described.

European AM Implementations

Accident management procedures and guidelines were developed in parallel with plant operation in Western Europe, except for the Sizewell B PWR, which had already implemented accident management during plant construction. Many Western European utilities selected the WOG SAMG approach as their primary vehicle for accident management for PWRs. This was done at most PWRs in Belgium (Tihange), Netherlands (Borssele), Spain (Asco, Almaraz, Jose Cabrera), and Switzerland (Beznau). Some other plants extended the range of their EOPs into the SAMG domain. This was done in the Belgium (Doel) and the UK (Sizewell B).

The French PWRs followed their own SAMG approach, which is not as detailed as the WOG SAMG approach and also implemented some hardware features (such as a separate RCP seal cooling system to prevent seal LOCAs during an accident and mobile cooling systems to fill an empty steam generator). At the German PWRs, an Emergency Manual was created to deal with accidents but with limited guidance (compared with US approaches) for the case of core melt. German PWRs also implemented hardware measures to reduce the likelihood of severe accidents (additional power lines, bunkered decay heat removal systems, qualification of relief valves for bleed & feed operation, and mobile water sources to fill an empty steam generator).

Only the Finnish PWRs at Loviisa implemented a fully integrated SAM plan including both hardware and software measures supported by a plant-specific research programme. Loviisa did not develop ex-vessel severe accident management guidance, but instead implemented changes to make it very unlikely ("physically unreasonable") that RPV failure would occur in a severe accident (a probabilistic approach, called Risk Oriented Accident Analysis Methodology, ROAAM).

European Regulators AM positions

Unlike the US situation, many Western European countries saw action by regulators to require accident management. Regulatory requirements were put forward in Belgium (directly to Tihange NPP, indirectly to Doel NPP), Finland, France, Germany (as a binding RSK recommendation), Netherlands, Sweden, Switzerland, and the United Kingdom (implicitly in the regulations on tolerable risk). No such requirements were issued in Spain as the Spanish regulatory authority follows the regulation of the country of origin of the plants (USA, Germany). Hence, in Spain SAMG was a voluntary action by the utilities. The presence of regulatory requirements does not automatically mean that severe accidents are in the licence basis - in most countries the regulator does not formally approve the SAMGs.

Coverage of accident scenarios

Most Western European SAMGs cover all phases of a severe accident, from in-vessel core degradation, RPV failure, basemat attack, hydrogen combustion, containment bypass, and so on. Finland, as noted above, does not consider the ex-vessel phase because it has been excluded as physically unreasonable by plant modifications to ensure in-vessel retention of core debris. The German approach, while it does not have full SAMGs in place (in the WOG SAMG sense), does consider various severe accident phenomena such as hydrogen generation and basemat attack. Only a very few NPPs in Western Europe are working on shutdown SAMGs (Borssele in Netherlands and Goesgen in Switzerland). Long term SAM provisions are considered within the scope of SAMGs only in Sweden.

European SAMGs Organisation

Western European SAMGs are split between those for which SAMGs are guidance and those for which the SAMGs are prescriptive procedures. The plants with WOG SAMGs clearly are among those, which employ guidance; the French PWRs also employ guidance. More prescriptive procedures are used in the German PWRs, at Trillo in Spain (which employs the same technology as the German PWRs), in Doel in Belgium, and at Sizewell B in the UK. Where accident management response remains with the control room operators, the SAM approach is prescriptive. Where accident management response shifts to the TSC, the SAM approach is in the form of guidance.

European Harmonisation Status

In view of the above, it is apparent that there is not formally a Western European practice or standard for accident management. The partners in the Severe Accident Management Implementation and Expertise project of the EU (SAMIME) [Vayssier 02] developed a consensus position on some basic aspects of SAM. All of the Western European countries with active nuclear power programmes were represented, along with organisations from Slovenia and Slovakia. The consensus position does not reflect the position of every partner in all particulars, but it provides the next best thing in terms of the Western European state-of-the-art in severe accident management. The consensus opinion consists of the following aspects:

- *The Need for SAM Written Guidance or Procedures:* There was a full and definite consensus that well developed and institutionalised, written guidance is a valuable tool for handling a severe accident and clearly should be available to support staff during the highly stressful conditions associated with this kind of event.
- *The Type of Guidance (Detailed Written Directives, Ranking of Potential Counter-measures, or Add-On to EOPs Directed at Restoration of Lost Safety Functions):* There was no clear consensus opinion on which of these options is best.
- *Coverage of SAM:* There was broad consensus that events beyond the design basis but before core damage is expected (e.g., where bleed & feed is employed), core damage but not vessel failure, and core damage with vessel failure and basemat attack should be covered by defined and written SAM guidance.
- *Mechanistic or Probabilistic SAM Basis:* The SAM guidance should be based on a mechanistic approach since it is hard to exclude scenarios based on probabilistic considerations unless it can be shown that phenomena are physically unreasonable or extremely unlikely. (French organisations considered this issue as not meaningful for accident management.)

- *Technical Basis for SAM:* There was a general consensus that there should be an appropriate technical basis for both the preventive and mitigative part of SAM guidance, the technical basis being the knowledge base from which the SAM guidance is developed, including the effect of planned operator actions on the degraded plant condition.
- *Flexible Guidance or Prescriptive Procedures:* Severe accident management could consist of flexible guidance or prescriptive procedures. There was a broad consensus that in the domain of the MCR, SAM mitigation measures responding to initiating events before establishment of the TSC should be prescriptive. Once the TSC is established, mitigation measures should be more of the guidance type, but once a strategy is selected the instructions to the operators should be clear and unambiguous.
- *Need for Plant Diagnosis:* There was general consensus that insights into the state of the core, vessel, and containment would be valuable (especially the core and containment), and that approaches not using such insights explicitly (such as WOG SAMGs) could be enhanced by such insights. There was also consensus that SAM guidance should be available where such insights are missing, in the form of default guidance.
- *Need to Address Potential Negative Consequences:* There was broad consensus that SAM guidance must address potential negative consequences of actions, and that this should be done before hand, as it is not possible during the stressful situation of an accident.
- *SAM Priorities:* There was a large consensus that the priorities for SAM should be defined. In the beginning of an accident, the priority is with core cooling. If the core is damaged, priority generally shifts to fission product boundaries. If fission product boundaries are intact and not challenged, operators again should try to terminate core damage progression. If the integrity of fission product boundaries is lots, priority should shift to minimizing releases. Where challenges are mixed (bypass), priorities should not be set before analyzing the situation.
- *Need for EOP Exit Criteria to SAMGs:* There was a consensus that exit criteria from the EOPs to SAMGs should be established because overall responsibility may shift from the MCR to the TSC or Emergency Manager, and because some of the actions in the SAM domain may conflict with actions in the EOP domain. Where EOPs are not exited, the priority of SAM actions and their consistency with the EOPs should be checked.
- *Need for Throttling and Termination Criteria:* There was broad consensus that it is useful to have throttling and/or termination criteria for SAM actions.
- *Responsibility for SAM Guidance Implementation in MCR or TSC:* There was a large consensus that responsibilities should be clearly defined, but apart from this the situation is quite open and decisions could be made by either the MCR or TSC, or there could be a shift from the MCR to the TSC. The method chosen should be consistent with the plant's existing philosophy on emergency response and with the applicable legal framework. Partners implementing US owners groups' SAM guidance felt that the responsibility remains with the TSC and the responsible Emergency Manager.
- *Separation Between Evaluation and Decision-Making:* There was a broad consensus that the chain of command should be clear, and that it should be clear who is performing assessment and who will finally decide what actions should be taken. There was no a priori preference whether there should be a clear distinction between these two groups of people. This is dependent on the structure of the emergency response organisation and the organisation for normal operation.
- *Availability and Use of Outside Guidance:* Outside technical support was generally welcomed, but the guidance should be so comprehensive that the available plant staff and management should be able to handle the events without such support at least for the first several (ten) hours of an accident. Some partners felt that if possible, external guidance should be provided and experts should be available as soon as possible.

- *Guidance for Equipment/System Restoration:* There was general consensus that guidance should be available to determine which systems could be best brought back into service. One should know the time to take critical actions (reach compartments, spend time in a hostile environment, repair components, build shielding, etc.), but there should also be room for ad hoc decisions.
- *Limitation of SAM Guidance to Existing Plant Capabilities:* The consensus was that plant modifications are primarily a tool to achieve a predefined safety level, including modifications for obtaining a meaningful severe accident management capability. Plant staff should therefore investigate their plant capabilities and eventually consider such modifications. Once having reached this safety level, plants should focus on how to handle a severe accident with their existing or improved capabilities. Limited further modifications (primarily for I&C) should still be considered for implementation if they can provide great benefit at reasonable cost. (Many plants have made modifications to cope with severe accidents, and their existing plant capabilities already include such provisions.)
- *Regulatory Involvement:* There was consensus that there should be regulatory involvement but at a lower level than in the design basis area. The role should be to define minimum acceptance criteria and assess the SAM guidance, together with the assessment of plant vulnerabilities. It is important that there is a consistent regulatory position with respect to the EOP domain and the SAM domain throughout the regulatory process (regulations, licenses, and regulatory oversight).
- *Training Interval:* There was broad consensus that after preliminary training, an interval of one to two years for refresher training is reasonable.
- *Decision Making Ultimately Onsite or Offsite:* There was broad consensus that the ultimate responsibility for decision-making should be with the plant management. Regulatory authorities should be informed (and in some countries must be included in the decision making process prior to actions which would severely impact the public, like venting the containment). The responsibility should be consistent with the emergency plan.
- *Processing of New Information:* There was consensus that new insights, as well as feedback from training, should be factored into the SAM guidance. Periodic revisions are therefore appropriate and should be done at such intervals that the SAM guidance remains a "living" tool.

Finally, the EU-project identified also issues, which were considered resolved, and candidate areas for further severe accident research:

The issues considered to be resolved include direct containment heating (DCH); steam explosion-induced containment failure, and global hydrogen combustion in PWR large dry containments that are not highly compartmentalized.

Candidates for further research were found to be ex-vessel fuel-coolant interactions, molten core-concrete interaction (MCCI), local hydrogen burns, retention of fission products in a water pool and in the steam generator secondary side, performance/development of I&C in severe accidents, and the effects of SAM strategies on Level 2 PSA.

ANNEX B

SEVERE ACCIDENT SCENARIOS CALCULATED FOR OR CONSIDERED IN THE PN7 PROJECT

Table B.1. SA scenarios calculated for Temelín NPP – Czech simulations with the MELCOR code [Sykora 01a]

Ref. No.	Sequence Conditions	Key Events	Threats to the containment integrity	Comments / conclusions on SAM measures
1.1 A	**Initiating Event:** PRISE 40 mm **Availability of systems:** HPI in service, HPR fails (at CONT sump level 0,5 m) LP ECCS and CSS - not available **Other assumptions:** No operator actions to stop the leak PAR and H_2 burning - modelled Hot leg thermal creep - modelled	End of HPI 7 343 s Core melt start 71 989 s RPV bottom failure 104 703 s H_2 burn in r. cavity 104 706 s End H_2 burn in C. 104 765 s Hot leg thermal creep not predicted (HP path)	CONT pressure increase after RPV failure (180 t of water from SITs and 50 t from ruptured SG) CONT temp <165 °C CONT pressure < 0,23 MPa Short term H_2 burning CONT not challenged	Considerable FP retention in the RCS before RPV failure Significant FP release after RPV failure
1.1 B	**Initiating Event:** PRISE 40 mm **Availability of systems:** as in 1.1 A, **Other assumptions:** Accident induced hot leg LOCA - not postulated RPV bottom failure under HP conditions Molten corium spread into CONT. DCH modelled	End of HPI 7 343 s Core melt start 71 989 s RPV bottom failure 104 787 s Start DCH RC door open End of DCH 104 797 s H_2 burn in RC 104 859 s End of H_2 burn in RC. 104 879 s	CONT pressure increase after RPV failure (180 t of water from SIT+ 50 t from ruptured SG) CONT pressure < 0,55 MPa DCH causes higher P, T than H_2 burn in 1.1A, parallel global H_2 burn + DCH not predicted. CONT integrity challenged due to MCCI after 5 days. MCCI very slow vertically, stopped in radial direction	Significant FP retention in the RCS before RPV failure due to counter current flow in intact loops. Most FPs are released after RPV failure.
1.2	**Initiating Event:** LB LOCA 200 mm (PRZR surge line) **Availability of systems:** ECCS – not available CFS – 2 SITs available CSS - in service EFW - not available **Other assumptions:** Reactor cavity open to the adjacent room GA302 PAR and H_2 burn - not modelled?????	End of SIT injection 400 s Beginning of core uncovery 650 s Beginning core melt 2 011 s RPV boiled dry 13 500 s RPV bottom failure 25 095 s Gas temperature maximum 3 000 K	CONT pressure <0,25 MPa (CSS in service) Release of corium LP path 190 t Corium pool surface 100 m² MCCI in radial direction stopped MCCI in axial direction 1 cm/h (after 90 h) and decreasing, Total depth of ablated basemat at one day ~1,2 m.	FP released to CONT washed down to the sump by CSS CONT intact, releases low Distribution of FPs in CONT: Sump 59% Corium 29% CONT surface 9% CONT atmosphere 3%
1.3	**Initiating Event:** LB LOCA 200 mm (PRZR surge line) **Availability of systems:** as in 1.2 **Other assumptions:** H_2 accumulation without burn assumed H_2 detonation assumed (supersonic wave velocity), H_2 detonation conditions: H_2 > 14%, O_2 > 9%, steam < 30% vol. Resulting CONT failure: hole in the wall, ~ 0,01 m²	End of SIT injection 400 s Beginning of core uncovery 600 s Beginning core melt 1940 s RPV boiled dry 14 000 s RPV bottom failure 25 590 s H_2 detonation and loss of containment integrity 25 650 s	CONT parameters before H2 detonation Pressure 0,115 MPa Temperature 323 K Pressure peak after H2 detonation 1,3 MPa H2 production continues afterwards from MCCI	FP released to CONT washed down into the sump by CSS Increased Iodine and Cs release following loss of CONT integrity. Discharge of noble gases 500 times higher than in 1.2

Ref. No.	Sequence Conditions	Key Events	Threats to the containment integrity	Comments / conclusions on SAM measures
1.4	**Initiating Event:** Station blackout (SBO) **Availability of systems:** Aactive safety systems (ECCS, CSS, EFW) - failed CFS - 4 SITs available **Other assumptions:** PAR + H_2 deflagration - modelled Corium pool restricted to the reactor cavity Corium melt through the instrumentation channels then concrete attack to basemat - modelled	First PRZR SV open 5 530 s Core uncovery starts 8 550 s Core melt starts 10 660 s Core dry 11 770 s RPV bottom failure (DCH, SITs injection) 33 874 s End of DCH 33 882 s Melt through into the instrumentation channels 34 282 s	Release of corium to the RC following RPV failure 146 t Corium spread directly into CONT due to DCH 32 t No significant corium cooling No steam explosion CONT integrity challenged (peak pressure ~ 0,83 MPa) MCCI in instrumentation channels stopped Basemat penetration down to 1,7 m 24 h (remaining thickness sufficient to hold corium weight & pressure load).	Containment integrity challenged by 0,83 MPa), but the likelihood of CONT failure small (0,8 MPa corresponds to 5% conditional probability of failure) CONT leak corresponds to design limits 0,1% vol/24 h Source term (% of core inventory): Noble gases 0,09% Cs 0,0009% Iodine 0,001%
1.5.	**Initiating Event:** LB LOCA 200 mm (PRZR surge line) **Availability of systems:** ECCS HPI – failed, HPI restored after RPV failure, ECCS LPI – one train restored after RPV failure CFS – 2 SITs available CSS, EFW - not in service **Other assumptions:** Reactor cavity open to the adjacent room GA302 PAR not functional, H_2 burn modelled.	End of SIT injection 410 s Beginning of core uncovery 450 s Beginning core melt 2570 s RPV bottom failure 25 903 s H_2 burn starts 25 913 s Water injection into RC 30 000 s Second H_2 burn 42 892 s Third H_2 burn 56 203 s Base mat failure > 5 d	RPV failure at LP H_2 released in-vessel phase 760 kg Series of H_2 burns observed CONT integrity not challenged. Corium pool cooled by ECCS flow (about 120 m^2) ECCS flow slows down significantly MCCI	CONT leak 0,1%vol/24 h Source term calculated assuming direct release to the environment Water layer over corium decreases source terms Aerosols not washed down to the sump (more released than in 1.2)

Table B.2. SA scenarios calculated for Temelín NPP – IRR/ARCS simulations with the MELCOR code (conducted by Pisa University)

Ref. No.	Sequence Conditions	Key Events		Threats to the containment integrity	Comments / conclusions on SAM measures
2.1	**Initiating Event:** PRISE 40 mm **Availability of systems:** All systems available at time of initiating event HPI in service, HPR fails (at CONT sump level 0,5 m) LP ECCS and CSS - not available (no inventory) **Other assumptions:** Operators isolate FW to affected SG (per EOPs) Operators fail to depressurize the RCS in the short term to terminate primary-to-secondary leak Operators fail to depressurize the RCS system in the long term, having established makeup to the sump Operators do not perform feed and bleed	40 mm PRISE Scram BRU-K operates HP injection TK injection (charging) SITs discharge EFW tanks empty Start of Core Heatup Start of H_2 Production Core Slump to Lower Head RPV Failure Basemat thickness at 1,4 m	0 s 400 s 400÷ 1000 s 600÷14 400 s 3 600÷18 800 s 35 000÷71 000 s 330 000 s 451 600 s 471 200 s 474 200 s 493 700 s 547 600 s	Containment bypass, primary-to-secondary leak, with BRU-A valve performing per design (opening and closing on demand) RPV failure at high pressure & DCH cause CONT pressure rise to 0,63 MPa CONT Hydrogen <6 vol.% after RPV failure	Mitigation of source term by FW injection fo affected SG per SAMGs Operators have 5 days to recovery injection source; core damage not credible due available time and onsite water sources This scenario will <u>not</u> be a severe accident sequence with implementation of SAMGs
2.2	**Initiating Event:** SBO **Availability of systems:** No active systems available Batteries deplete in 1 hour PORV assumed unavailable EGRS assumed unavailable **Other assumptions:** No recovery of AC power	Loss of Offsite Power Failure of All Diesels Failure of EGRS & PORV BRU-A opening Loss of DC power Dryout of SGs Start of Core Uncovery Start of H_2 Production H_2 Released to Containment Core Slump to Lower Head RPV Failure Penetration of Serpentine Basemat Thickness at 1 m	0 s 0 s 0 s 200 s 3600 s 7100 s 10 300 s 14 700 s 14 800 s 19 800 s 20 590 s 27 900 s 43 650 s	DCH causes pressure <0,5 MPa CONT pressure and Basemat penetration trending toward containment failure in same time frame (>114 000 s) CONT steam inerted (at 11 000 s) well before RPV failure (at 20 590 s) and remains steam inerted until overpressure failure or basemat failure CONT hydrogen remains below 4 vol.% due to long term depletion of hydrogen by PARs	Sequence deliberately modeled as unmitigated, high pressure sequence (PORV & EGRS assumed unavailable) to test robustness of containment to DCH - no threat of containment failure from DCH (pressure < 0,5 MPa) Threat of containment failure due to basemat penetration or overpressure at 31,6 h (1,3 d) without SAMG actions or AC power recovery SAMG action to inject water from fire trucks to containment could delay basemat penetration if injection occurs early Recovery of AC power by 32 hours is <u>extremely</u> likely (98%), and would allow mitigation of overpresure by spraying or venting

Ref. No.	Sequence Conditions	Key Events		Threats to the containment integrity	Comments / conclusions on SAM measures
2.3	**Initiating Event:** SB LOCA **Availability of systems:** Common cause failure results in unavailability of HPI, LPI, and CSS TK system available (charging pumps) **Other assumptions:** Operators depressurize the RCS using the PORV per EOP and SAMG guidance	Small LOCA (50 mm) Reactor scram BRU-A opening TK (charging) injection SIT discharge Start of core heatup Start of H2 production Core Slump to Lower Head RPV Failure Penetration of Serpentine Basemat Thickness at 1 m	0 s 2 s 50 s 3 600÷18 800 2 900÷11 500 21 350 26 150 27 200 30 000 33 900 60 000	No DCH threat - operators act to depressurize RCS in accordance with EOPs and SAMGs H_2 combustion threat calculated with GASFLOW - there is a sensitive cloud formed for several hundred seconds if the break occurs in the SG box without the pressurizer (but only for a higher H_2 source; calculation based on 0,8 kg/s peak source rate) CONT overpressure threat after 15 hours BASEMAT penetration threat after 30 hours	No DCH threat due to operator action to depressurize RCS Energetic H_2 combustion threat exists for short time in SG box; conservative probabilistic evaluation shows very low risk SAMGs provide for containment venting to mitigate overpressure threat SAMGs provide for alternate water sources for debris cooling to delay basemat failure timing
2.4	**Initiating Event:** PRISE 40 mm **Availability of systems:** Same as 2.1, but BRU-A valve on affected SG assumed to fail open on first opening HPI in service, HPR fails (at CONT sump level 0,5 m) LP ECCS and CSS - not available (no inventory) **Other assumptions:** Same as 2.1	40 mm PRISE Scram BRU-A on affected SG opens HP injection TK injection (charging) done SITs discharge start Start of Core Heatup Core uncovered RPV Failure	0 s 120 s 822 s 600÷8 500 s 15 000 1 200 36 980 59 500 67 570	Containment bypass, primary-to-secondary leak, with BRU-A on affected SG stuck open H_2 combustion not a threat because about half of the hydrogen is released outside containment due to the bypass, and the remainder is released at the time of RPV failure and rapidly mixed with the containment atmosphere to yield a low concentration DCH threat limited because H_2 contribution is lower	No DCH threat due to operator action to depressurize RCS in accordance with SAMGs RCS depressurization in combination with makeup to the sump from 3×400 m^3 borated water tanks per SAMG strategy could recover sequence before core damage (about 4 hours available from complete loss of injection sources to start of core heatup) SAMGs direct operators to fill affected SG with feedwater to mitigate PRISE FP release path. Without SAMG strategy implementation, sequence would be a core damage sequence with potential for large release via containment bypass

Table B.3. SA scenarios calculated for Temelín NPP – ENCONET's simulations with MELCOR code (conducted by ENPROCO)

Ref. No.	Accident sequence conditions	Key events and their timing	Threats to the containment integrity	Comments / conclusions on SAM measures
3.1	**Initiating Event:** SB LOCA 15 mm **Availability of systems:** Loss of HPIS, EFWS, and TK system, SITs and LPIS available. **Other assumptions:** SAMG entry point – core exit temp 650 °C. RCS depressurisation using EGRS & PORV EGRS peak flow rate - 30 kg/s.	Core uncovery starts 1,2 h EGRS lines from PORV and RPV open 2,88 h Pressure drops & SIT Injection starts 3,38 h LPIS on (periodic injections) 6,51 h PORV opened. 14,1 h	No RPV failure. Mass of H_2 generated ~390 kg. No hazard of containment failure. Safe stable state achieved.	SAM measures (RCS depressurisation and LPI) successful (given a high EGRS flow rate)
3.2	**Initiating Event:** SB LOCA 15 mm **Availability of systems:** As above **Other assumptions:** SAMG entry point – core exit temp. 650 °C RCS depressurisation using EGRS	Core uncovery starts 1,2 h EGRS lines open 3,16 h SITs inject & PORV opened. 4,4 h	No RPV failure No hazards to containment	For the high flow rate assumed SAM measures (depressurization using EGRS and PORV) successful.
3.3	**Initiating Event:** SB LOCA 15 mm, **Availability of systems:** Loss of EFWS, HPIS, LPIS available, **Other assumptions:** SAMG entry point core exit temp. 650 °C. RCS depressurisation using EGRS & PORV Flow rate through EGRS reduced to 14 kg/s	Peak flow rate through EGRS insufficient, RCS pressure above HPIS delivery head, Core fully uncovered (110 kg H_2 produced) 3,3 h PORV opened 3,75 h	No RPV failure, No hazards to containment. No PRV failure	Owing to opening of PORV SAM successful. No PRV failure, so no hazard of MCCI. Containment intact.
3.4	**Initiating Event:** LOFW, **Availability of systems:** ECCS available, TK not available **Other assumptions:** RCS depressurisation using BRUA, EGRS & PORV BRUA opened at 0,5 h EGRS lines and PORV opened when core exit temperature 650 °C,	SG empty 1 h Core uncovery begins 2,13 h RCS depr. starts (by operator) 2,36 h HPIS injection starts 2,55 h SIT injection starts 2,84 h Core covered, fuel cooled.	Gap inventory released from all fuel rods, Hydrogen mass released negligible (~ 5 kg), Safe stable cooling state achieved.	SAM successful. The capacity of PORV supported by EGRS is sufficient to depressurize the RCS and bring situation under control

Ref. No.	Accident sequence conditions	Key events and their timing	Threats to the containment integrity	Comments / conclusions on SAM measures
3.5	**Initiating Event:** LOFW, **Availability of systems:** ECCS available, TK not available, **Other assumptions:** BRUA not opened EGRS and PORV opened as in 3.4 (above)	Core uncovery starts 2,34 h RCS depressurization starts 3,25 h HPIS starts to inject 3,41 h SITs start to discharge 3,59 h Core covered again 3,65 h	Gap inventory released from all fuel rods. Hydrogen generation negligible (~1,4 kg). No hazard to containment. Safe stable cooling state achieved.	SAM successful
3.6	**Initiating Event:** LOFW **Availability of systems:** ECCS and EFWS lost, **Other assumptions:** No SAM measures, Updated concrete composition (as given by Czech side)	SGs empty 2,5 h Core uncovery / relocation start 3,6 h Core uncovery complete 4,5 h H_2 generation in-vessel phase 640 kg RPV bottom head fails at 7 h Basemat penetration 39 h	Vigorous initial MCCI, CONT pressure <0,4 MPa. CONT pressurisation stops below design pressure. H_2 generated due to MCCI - In serpentinite (up to 16 h) ~ 850 kg - In base concrete (16 to 42 h) ~1 120 kg CONT atmosphere inert - Due to high steam concentration since 7 h - Due to lack of oxygen since 10 h	MCCI ablation rate: - in serpentinite 7,7 cm/h - in base concrete 11,9 cm/h .
3.7,	**Initiating Event:** LB LOCA 200 mm break **Availability of systems:** Active ECCS – unavailable CFS – 4 SITs ??? **Other assumptions:** No SAM actions	SIT injection 220÷450 s H_2 generation started 1 700 Core support plate fails 3 232 Lower support plate fails 8 800 Bottom head fails 15 150 Serpentinite penetrated 50 000 Basemat penetrated 171 000	H_2 released from RPV 130 kg H_2 produced due to MCCI - In serpentinite 1 140 kg - In base concrete 1 810 kg - average ablation rate 7÷8 cm/h Max. recombination rate of 1 PAR 1 g/s Containment inert, pressure < 2,8 Bar	After 48 hours the basemat penetrated. At that time activities of fission products are still high because CSS is not in operation.
3.8	**Initiating Event:** LB LOCA 200 mm break **Availability of systems:** Active ECCS - unavailable CSS recovered at 16 hours **Other assumptions:** No SAM actions	As above, CONT pressure increased to 0,5 MPa (after CSS recovery) due to hydrogen deflagrations	Containment remains safe in spite of hydrogen deflagration	With CSS actuated the activities of FP releases rapidly decrease, about 100 times
3.9	**Initiating Event:** LB LOCA, **Availability of systems:** Active ECCS - unavailable **Other assumptions:** Corium spreading over 100 m^2 by opening RC door, Cavity door opened before RPV failure, Top cooling of corium with water considered optional.	As above till RPV failure (i.e. 15 150 s) Basemat penetration ~37 h	H_2 concentration up to 50 h below 10% Oxygen concentration after 15 h below 5%. CONT peak pressure (in case of CSS actuation) 6,15 Bar Increased generation of gases due to MCCI, the ablation rate decreases in case of debris top cooling.	With CSS actuated the activities of FP releases rapidly decrease, about 100 times

Table A.4. SA scenarios calculated for VVER 1000 NPPs – TACIS results [Schoels 02a, Schoels 02b, Schoels 02c]

Ref. No.	Sequence Conditions	Key Events	Threats to the containment integrity	Comments / conclusions Proposed SAM measures
4.1	**Initiating Event:** LB LOCA 850 mm + SBO (Balakovo NPP) **Availability of systems:** Active ECCS – not available CSS – not available SG feed water – all systems lost **Other assumptions:** Calculations performed with ESCADRE code	Core degradation, corium formation, and release from RPV start 0,55 h Basemet melt through rate: - Serpentinite layer (1,34 m) 1,3 d - Silicate base concrete (1,3 m of 2,4 m) 30 d FPs released from the core – about 43% of gaseous and volatile before beginning of corium formation	Maximum CONT pressure 0,425 MPa Maximum CONT temperature 148 °C Pressure peak in SG box after RCS break 0,53 MPa After RPV failure 0,355 MPa then decreases, Maximum CONT temperature 505 °C (Calculation with JERICHO code) CONT integrity challenged (MCCI) > 30 d	CONT leaktightness kept. In the first day only 0,09 kg of aero-sols and 2 E-6 kg of iodine released to the atmosphere (at 0,3%/day containment leakage rate). Practically all mass of aerosols (99,3%) is deposited on the containment surfaces [Schoels 02a]
4.2	**Initiating Event:** MB LOCA 80 mm + SBO **Availability of systems:** AC power supply – all sources lost **Other assumptions:** Calculations performed with ESCADRE code	Core degradation, corium formation, and release from RPV start 2,5 h Basemat melt through rate: - Serpentinite layer (1,34 m) 1,3 d - Silicate base concrete (1,3 m of 2,4 m) 30 d FPs released from the core – about 3% of gaseous and 2,5% of volatile before beginning of corium formation	Maximum CONT pressure 0,515 MPa Maximum CONT temperature 260 °C Total H_2 generated due to MCCI 1 586 CONT integrity challenged (MCCI) > 30 d	CONT leaktightness kept. Total< mass of aerosols released from the core is 175 kg, but after 1 day only 1,78 kg is in suspended state and after 2 days only 0,13 kg. That means that 99,99% of all aero-sols are deposited on containment surfaces within 2 days [Schoels 02a]. In the first day only 0,044 kg of aerosols and 1E-6 kg of iodine released to the atmosphere (at 0,3%/day CONT leakage rate).
4.3	**Initiating Event:** SBO **Availability of systems:** Loss of all AC sources, failure of active ECCS **Other assumptions:** Calculations with MELCOR	Core dried out, 6 500 Fuel cladding fails 9 080 RPV failure 19 000 SITs injection partly freezing the core Corium falls into reactor cavity 23 500 Basemat (2,4 m thick) failure 81 500	H_2 generated - in-vessel phase 332 kg H_2 generated due to MCCI 2 620 CONT integrity challenged (MCCI) > 22,6 h	The vertical erosion of concrete calculated by ESCADRE is nearly 2 times lower than by MELCOR (1,3 versus 2,4 m). The authors indicate that the ESCADRE is more reasonable

* *Application of stratified corium model of melting through serpentine concrete and mixed corium model for silicate base concrete in WECHSL code shows that the basemat would not be penetrated within 30 days. However stratified model indicates basemat melt-through times of 10,27 days for LB LOCA and 6,63 days for SB LOCA, and mixed model both for serpentine and silicate concrete 23 days for LB LOCA and 12,7 days for SB LOCA. The authors conclude that the time of basemat full melt-through (without corium spreading) is from 10 to 23 days for LB LOCA and from 6 to 13 days for SB LOCA [Schoels 02a].*

ANNEX C

LIST OF AUSTRIAN PROJECTS

AUSTRIAN PROJECTS IDENTIFICATION

PN 1	Severe Accidents Related Issues – [Item No. 7a] *
PN 2	High Energy Pipe Lines at the 28,8 m Level (AQG/WPNS country specific recommendation) [Item No.1] *
PN 3	Qualification of Valves (AQG/WPNS country specific recommendation) [Item No.2] *
PN 4	Qualification of Safety Classified Components [Item No. 5] *
PN 5	Regular bilateral Meeting 2002
PN 6	Site Seismicity [Item No. 6] *
PN 7	Severe Accidents Related Issues – [Item No. 7b] *
PN 8	Seismic Design
PN 9	Reactor Pressure Vessel Integrity and Pressurised Thermal Shock [Item No. 3] *
PN 10	Integrity of Primary Loop Components – Non Destructive Testing (NDT) [Item No. 4] *
PN 11	Regular bilateral Meeting 2004

** The Items are related to Annex I of the “Conclusions of the Melk Process and Follow-up”*