7 Cluster "Containment Integrity in Severe Accidents"

Issue 1: Containment Bypass & PRISE Accidents
Issue 4: Containment Design & Arrangement
Issue 5: Probabilistic Safety Assessment (PSA)
Issue 16: Hydrogen Control
Issue 26: Beyond Design Basis Accidents (BDBAs)

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7.1 Introduction

Five problem areas have been identified related to containment integrity in severe accidents. These problem areas can be grouped under the headings of (a) phenomenological threats to containment integrity, and (b) containment bypass:

(a) Phenomenological threats to containment integrity

— Elimination of the hydrogen problem
— Minimising the potential for high pressure core melt ejection
— Minimising the potential for reactor cavity melt-through
— Minimising the potential for late overpressure failure of the containment

(b) Containment bypass
7.2 Identified Problems and Solutions

7.2.1 Phenomenological Threats to Containment Integrity

7.2.1.1 Hydrogen Problem

Identified Problem

Generation of hydrogen is an unavoidable outcome of severe accidents in the Temelín design. A system of passive autocatalytic recombiners (PARs) has been installed in the containment, however this system is sized and placed in locations corresponding to design basis accident conditions rather than severe accident conditions. It is possible under severe accident conditions for hydrogen detonation to pose a threat to containment integrity.

Solution to the Identified Problem

Solutions are possible in the following areas:

- Analyse the placement and efficiency of the PARs under severe accident conditions using an appropriate simulation model, which can adequately model natural convection processes inside the containment.
- Upgrade the system of passive autocatalytic recombiners (PARs) to severe accident conditions. This may require an increase in the number of PARs and/or placement of PARs in different locations in order to avoid conditions conducive to hydrogen detonation.

7.2.1.2 Potential for High Pressure Core Melt Ejection

Identified Problem

Some severe accident sequences are associated with primary system pressure at the time of reactor pressure vessel failure sufficient to result in high pressure ejection of core melt debris from the reactor vessel. Such an outcome can result in direct containment heating, large differential temperatures between the containment liner and the concrete containment walls causing the liner to leak, and/or direct contact between core melt debris and the containment liner, leading to liner failure. Significant liner leakage or failure can result in a containment leak rate much greater than the design basis leak rate of 0.1 volume percent per day.

Solution to the Identified Problem

Solutions could be possible in the following areas of improvement:

- Installation of a system for intentional (manual) depressurisation of the primary system to reduce pressure in case of core melt accidents.
- Analysis of the mechanical consequences to the reactor cavity and other containment structures of core melt release under primary system pressures higher than containment pressure.
- Analysis of the thermal consequences of core melt release into the containment under high primary system pressure, in particular the thermal response of the containment liner.

7.2.1.3 Potential for Reactor Cavity Melt-Through

Identified Problem

Unimpeded interactions between molten core debris and the reactor cavity (including neutron gauge tubes in the cavity walls) can result in melt-through of the reactor cavity and release of core debris and airborne radioactive materials into the reactor building and subsequently into the atmosphere.

Solution to the Identified Problem

Improvements could be possible in the following areas:

- Detailed analysis of the pressurisation of the reactor cavity at vessel failure, with the aim to modify the reactor cavity door in a way that it opens (with high confidence) and allows spreading of the core debris out of the reactor cavity also under low pressure sequences (opening of the door under high pressure sequences is assured due to pressure differential).
- Evaluation of possible means for core debris cooling.

7.2.1.4 Potential for Late Overpressure Failure of Containment

Identified Problem

Continued pressurisation of the containment in a severe accident over the span of a few days can result in overpressure failure of the containment.

Solution to the Identified Problem

Solutions could be possible in the following area of improvement:

- Evaluate the possibility to install a containment filtered venting system. (Filtered containment venting systems are installed in pressurised water reactors in France, Germany, the Netherlands, and Sweden within the European Union, as well as in PWRs in Switzerland.)

7.2.2 Containment Bypass

Identified Problem

Contrary to the state-of-the-art in Member States of the European Union, certain severe accidents leading to containment bypass (steam generator tube rupture and steam generator collector leakage) at Temelín are dominant contributors to core damage frequency. There is insufficient proof that the frequency and consequences of these containment bypass accidents have been reduced to acceptable levels.
Solution to the Identified Problem

Solutions could be possible in the following areas of improvement:

- Steps to reduce the likelihood of steam generator tube rupture (SGTR) involving confirmation of the adequacy of steps to avoid primary water stress corrosion cracking (PWSCC) and the adequacy of loose parts control. (Historically, PWSCC and loose parts have been important contributors to SGTR events worldwide.)
- Provision of means to make additional water sources available to the containment sump in order to provide more time to respond to containment bypass accidents.
- Enhanced in-service inspection (ISI) of steam generator tube integrity and steam generator collector head restraining devices (which limit the leak rate in steam generator collector leak from 100 to 40 mm equivalent).
- Use of an automatic control system to replace human actions in responding to containment bypass accidents (such systems have been installed in PWRs in Germany).
- Incorporation of provisions into emergency operating procedures (EOPs) and severe accident management guidelines (SAMGs) to add secondary feedwater as necessary to ensure enhanced retention capabilities against fission product transfer from the steam generators to the atmosphere in containment bypass severe accidents involving steam generator tube rupture or steam generator collector leakage.

7.2.3 Timeline

All the necessary analyses for Unit 1 could be completed within one year; the extent and the time needed for procedural and/or physical modification measures resulting from the analyses cannot be estimated without knowledge of the results of the relevant analyses. State-of-the-art in EU member states, for a PWR being licensed in the late 1990s or later, would require resolution of hydrogen control (severe accident analysis and placement and number of PARs), high pressure melt ejection (depressurisation capability and related EOPs/SAMGs), containment overpressure failure (filtered venting), and containment bypass issues before fuel load. State-of-the-art in EU member states does not provide for a license to be granted to a PWR where reactor cavity melt-through could take place into an unfiltered compartment above grade.

7.3 Deviation from State-of-the-Art and Significance

Severe accidents are explicitly addressed in the 1988 and 1995 consensus documents on the safety of European Light Water Reactors /1-3/.

Without stressing the expectations mentioned in the 1995 consensus document with respect to new power plants (e.g., the EPR), reference is made to some requirements for existing plants. These requirements correspond to the results of major international activities devoted to resolving certain issues which in essence dominate the consequences of severe accidents.

In the context of containment bypass accidents, reference is made to the goal to minimise the frequency of such accidents so that they are not dominant contributors to the likelihood of severe accidents /1/. Reference is also made to the international efforts to understand and evaluate the consequences of High Pressure Melt Ejection and Direct Containment Heating /4/, the basic requirements to cope with the potential consequences
of the formation of hydrogen unavoidable under all severe accident conditions /5/ and the possibilities to mitigate large amounts of molten core material /6/.

There is general consensus that the conditional probability of containment failure due to phenomenological causes should be minimised by suitable severe accident management procedures addressing amongst other the following items:

- controlled reactor coolant system depressurisation to minimise the likelihood of high pressure melt ejection (HPME) /4, 6/;
- improved hydrogen control to prevent reaching explosion limits that would threaten the containment integrity /5/;
- limitation of corium concrete interaction to minimise the loss of integrity of the containment base plate /7/; and
- the use of filtered venting to avoid containment overpressurisation /1/.

With respect to the Environmental Impact Assessment for Temelin NPP reference is made to the Council Directive 89/618/Euratom /8/ on informing the general public about health protection measures to be applied and steps to be taken in the event of a radiological emergency, in particular Annex I, and the Commission Communication on the implementation of this directive /9/.

In terms of the criteria used to assess the significance of the deviation from the state-of-the-art relevant in EU Member States, the following comments are made:

- **Hydrogen Problem** — This deviation is classified as one of high safety significance due to the potential for early containment failure or early high leakage rates; moderate to large deviation in terms of cost to implement solutions.

- **Potential for High Pressure Core Melt Ejection** — This deviation is classified as one of high safety significance due to the potential for early containment failure or early high leakage rates; minor deviation in terms of cost to implement solutions and likely short implementation schedule because the solution, unless it involves installation of additional depressurisation capability, will at most involve changes to EOPs and SAMGs.

- **Potential for Reactor Cavity Melt-Through** — This deviation is classified as one of high safety significance due to the potential for containment failure and the forced abandonment of the reactor building — including the main and emergency control rooms — due to high radiation dose fields; moderate deviation in terms of cost to implement solution.

- **Potential for Late Overpressure Failure of the Containment** — This deviation is classified as one of high safety significance due to the potential for containment failure; large to moderate deviation in terms of cost to implement solution, which will be dependent on the type of filter selected, whether a new containment penetration is required, and what sorts of controls are placed on the use of the system.

- **Containment Bypass Accidents** — This deviation is classified as one of high safety significance due to containment bypass and the size of the unmitigated source term;
moderate deviation in terms of cost to implement solutions, i.e., less than "tens of millions of Euros or more", although it may take several years to implement an automatic control system.

Analysing the above issues in an appropriate manner will also have consequences for matters addressed in other issues, such as SAMGs and EP/EPZs.

7.4 Technical Arguments

7.4.1 Phenomenological Threats to Containment Integrity

7.4.1.1 The Hydrogen Problem

Under all accident sequences selected to study the containment behaviour for the Temelín NPP, inevitably large amounts of hydrogen would be produced due to physicochemical interactions between core fuel, steam-water mixtures, involved core structures, and last but not least that originating from core-concrete interactions.

The IAEA issue book for WWER-1000 NPPs, as one of the "main insights", cites hydrogen removal as an unsolved problem for WWER-1000 NPPs for both design basis accidents and severe accidents, and recommended remedying the problem as soon as possible as "a matter of safety significance" /25/.

Temelín is provided with twenty-two passive autocatalytic recombiners (PARs), the positioning of which is not known in detail to the Austrian side. (The Temelín POSAR indicates the containment subcompartments in which the PARs are located, but not their location in detail.) It is recommended to set up an analytical simulation model based on an appropriate code version application to simulate natural convection processes inside the containment. Anticipating that the positioning of the recombiners within the containment corresponds to the state-of-the-art to deal with hydrogen release in case of a design basis accident, such a model could serve to evaluate the efficiency of the PARs for the selected accident sequences. Supplementary guidance on how to proceed with this task can be found in a recent State-of-the-art Report (SOAR) for Containment Thermal-hydraulics and Hydrogen Distribution /12/.

In view of the strength of the Temelín containment, the primary goal of the proposed analyses should be to eliminate the potential for hydrogen detonations for the scenarios under consideration. If 22 PARs would not be sufficient, more recombiner units could be installed without delay as such devices are readily available. (Taking into account 3000 MWt as a first criterion for the hydrogen formation potential, and a free volume of 70,000 m³ as a basis for the distribution process, we estimate 30 to 45 recombiner-units may suffice.) If possible, the potential for deflagration and/or accelerated flames through self ignition by overheated recombiners should be minimised to avoid a difficult evaluation of consequences of hydrogen combustion for containment liner integrity.

The modeling concept (chosen nodalisation, use of subroutines incorporated within the code), in particular for the thermal-hydraulic containment simulation, were orally described at the expert meeting in Prague on 4 April 2001. The entire free containment volume of 67,000 m³ was simulated by a 2 volume configuration using the STCP code, and by a 5 volume configuration using the MELCOR 1.8.3 code version (this nodalisation is confirmed in the Czech Government report /24/). The simulation model was intended to cover the thermal-hydraulic conditions within the containment, possible hydrogen
combustion processes, the efficiency of installed recombiners, and last but not least the initial conditions for a subsequent possible hydrogen detonation.

For some selected analytical cases, hydrogen combustion was arbitrarily calculated to take place at the moment a pre-selected level of hydrogen concentration has been reached within a thermal-hydraulic control volume. For some other cases, obviously this option was not activated. The analytical simulation of the recombiner efficiency was not described in detail. Presumably recombiners have been simulated as lumped-together hydrogen sinks independent from possible local flow properties at the real location of the recombiners. This modeling concept is not at all suited to simulate the effects of self-ignition of passive recombiners and/or of the variety of different combustion modes upon ignition starting from natural convection-driven case mixture concentration levels.

The removal rate of a recombiner depends on the local hydrogen concentration at the recombiner position, which in turn is largely determined by the global natural convection-driven flow pattern in the containment. If high quality of prediction is required, the application of 3-D computational fluid dynamic codes is recommended in general. In certain cases, however, the application of a lumped parameter approach with an adequate spatial resolution of the free volume may be sufficient. A simulation concept as shown at the meeting on 4 April 2001 is far below minimum requirements experienced from the analyses of relevant containment experiments /12-15/.

The distribution of radioactive material between the involved solid and/or molten core structures and the gas phase into the containment was calculated by the available subroutines of the MELCOR code. The adopted modeling concept was supposed to cover the prediction of airborne radiological species available for transfer to the environment via anticipated leak paths from the containment to the environment. Basically it was assumed that the design normal leak rate would be determining the transfer of available airborne radioactive isotopes from the containment into the environment. A parametric variation of a steady-state leakage by a factor >1 supplemented the latter assumption to specify some source terms for the determination of the emergency planning zones (EPZs) for the Temelín NPP.

In summary, the applied modeling concept for Temelín NPP implies several serious shortcomings with respect to the envisaged simulation of containment phenomena and processes expected under severe accident conditions. Accordingly, the presented concept and the shown calculation results must be considered as not applicable to serve as a useful analytical basis for determination of EPZs and the associated administrative emergency measures.

The main reasons leading to this evaluation are summarised as follows:

1. At best, the nodalisation concept chosen with either MELCOR 1.8.3 or STCP may allow to predict the global pressure inside the containment provided that no hydrogen combustion takes place.

2. The nodalisation of the free volume of the containment with only 5 thermal-hydraulic balance volumes is not suited to properly predict natural convection processes in a 67,000 m³ containment. This is well known from the results of several analyses for gas distribution experiments, in particular from some blind pre-test predictions performed within the frame of International Standard Problems /14-15/. Even a
simplified combustion analysis would require the prediction of initial thermal-hydraulic conditions with a nodalisation concept with 80 up to 120 nodes to take into account natural flow convection in a containment with a comparable free volume /13/.

3. Without accurate simulation of the distribution of hydrogen, air, and steam preceding the moment of ignition, any prediction of combustion modes (deflagration, accelerated flames, transition of a deflagration into a detonation) will be completely misleading. Hence, the presented information on the occurrence and the effects of a possible hydrogen combustion or detonation are not applicable for the cases studied by Temelín NPP.

Some remarks made in the Czech Government report of March 2001 bear final comment. The report states that at the time when the reactor vessel bottom head fails, the hydrogen concentration in the containment is above the hydrogen ignition limit, but that ignition does not occur because of "the high steam volume in the containment atmosphere" /24/. The report does not indicate which accident sequence is modelled by this statement, but it clearly is not one of the dominant primary to secondary leakage accidents because there is no pathway for steam to reach the containment in these accidents; the steam is lost to the secondary side of the plant before the core melts and the vessel fails. Under such circumstances, hydrogen combustion would appear to be a possibility that cannot be excluded, especially so because the rate at which hydrogen would be added to the containment would by far exceed the capabilities of the hydrogen recombiners to accommodate.

The Czech report also cites operation of the containment sprays as reducing the cesium and iodine source term by a factor of 4 to 6 /24/. This is clearly not the case for the dominant accident sequences identified in the PSA — these sequences are containment bypass sequences in which the water supply for containment sprays is lost to the secondary side of the plant before core melt occurs. Thus, the sprays would not be available for source term attenuation as claimed for these dominant accident sequences. (They would also be unavailable in other sequences due to other causes. Indeed, the PSA suggests that for non-bypass sequences, containment sprays are not available for most accident sequences. This is evident from the containment event tree quantification in the PSA, which includes a specific top event for availability of containment sprays.)

Regarding other statements in the Czech report for hydrogen combustion (no detonation, leak rates resulting from hydrogen combustion, etc.) these statements are not regarded as well founded because of the modeling limitations discussed above. Similarly, statements in the Czech report regarding "deliberate hydrogen ignition" are not regarded as reasonable since no deliberate ignition system exists at Temelín.

7.4.1.2 The Potential for High Pressure Core Melt Ejection

The analytical results presented at the 4 April 2001 meeting for several scenarios show high pressure melt ejection to occur at primary system pressures well above 5 MPa. Due to the compartment configuration below the reactor pressure vessel, this would result in asymmetric core melt blowdown into containment regions adjacent to the reactor cavity below the reactor pressure vessel. For such situations, a simulation model must be able to predict the spatial temperature distribution fields, in particular those created in the liner of the containment by direct containment heating (DCH) effects. This would be necessary to assess the stability of the connection between liner and the containment concrete wall. (The IAEA issue book for WWER-1000 NPPs does not address this issue /25/.)
HPME sequences lead to a rapid transfer of energy into the containment atmosphere causing sharp non-uniform temperature rise in containment areas associated to those compartments connecting to the reactor cavity. Rapid heatup of the steel liner in those areas will follow, while the heat flow to the concrete is limited due to the differences in conductivity of the steel liner and the concrete. Large temperature differences between the involved liner areas and the concrete all connected by studs will cause localised expansion of the liner and considerable thermal stresses. Under such conditions, failure of the liner cannot be excluded. Additionally, local transient temperature information is also deemed to be necessary for an assessment of the thermal and mechanical behaviour of containment penetrations involved by DCH. The potential for a prompt loss of leaktightness must be taken into account.

Obviously, the potential for high pressure melt ejection (HPME) should be drastically reduced below the adopted threshold suitable for the emergency planning zone (EPZ) process, i.e., below approximately $10^{-7}$ per year. This should be achieved within a time period to be discussed by introducing deliberate primary system depressurisation into the SAMGs as already achieved in other countries /6/. At least the potential for deliberate depressurisation should be studied immediately to evaluate the possibilities and limits imposed by the existing design of Temelín.

As a consequence, a careful best-estimate analysis of the thermal-hydraulic effects of core melt behaviour under the remaining elevated coolant system pressure at the moment of melt release from the pressure vessel will be necessary to assess the impact on possible leak path formations (the German Risk Study Phase B identified this event tree as the ND*-sequence).

Without careful thermal analyses of the liner, severe damage cannot be excluded. This would essentially impede the overall leaktightness of the containment. For the time being, the assumption of leaktightness corresponding to the confirmed design conditions is not supported by the Czech presentations on the 4th of April.

Referring to answers on questions during the presentations in Prague on 4 April 2001, the dynamic response between the reactor pressure vessel and its immediate structural environment on HPME has not been taken into consideration. Corresponding forces acting on surrounding walls and on the bottom plate, as well as thrust forces acting on the reactor pressure vessel structures, will be considerable and should at least be analysed for the HPME cases /16/.

The Czech report from March 2001 mentions HPME and DCH at one point /24/, but does not discuss it beyond the mere mention of the phenomena. Similarly, other than to state that DCH was analysed using the CONTAIN code, the Nuclear Research Institute Řež plc report from March 2000 does not discuss the issue /10/.

7.4.1.3 The Potential for Reactor Cavity Melt-Through

The reactor cavity is a compartment, under and partially surrounding the reactor vessel, into which core melt would first be released after the reactor vessel fails. There is a pair of doors in a corridor leading from the reactor cavity to the lower area of the containment. These doors are normally closed.
The containment spray system, which includes a series of ring headers to which the spray nozzles are attached, is located in the dome of the containment. The spray water cannot reach the reactor cavity. In addition, the cavity floor is above the containment water level in all cases of LOCAs and emergency coolant injection. Thus, the arrangement at Temelín is properly referred to as a "dry cavity" design.

As noted in the 1995 Temelín PSA, the floor area of the reactor cavity is small compared with the heat contained in the core debris (and the heat generated during chemical reactions between core debris and structural materials in the reactor cavity). Unless the core debris can escape the reactor cavity area and spread out over a larger area, it will not form a coolable debris bed, and will thermally and chemically attack the floor structures in the reactor cavity. The bottom of the containment is gradually ablated until the thickness of the cavity reaches a point where it can no longer support the structures above, and the cavity fails (this point was given by Czech experts in the 4 April 2001 meeting as one meter of remaining thickness).

The time required to melt through the reactor cavity is uncertain, and it depends on numerous variables and assumptions in the analysis. But the uncertainty lies in the range of 18 hours up to 96 hours. When the cavity floor fails, core debris is deposited in a compartment below and outside the containment. In addition, if the containment is pressurised (as would be expected unless the containment leaks excessively or fails due to other causes before the cavity melts through), the containment atmosphere will blow down into the reactor building. (The ultimate structural capacity of the containment was given in the PSA as 1.1 MPa; at a higher pressure, the containment is likely to fail.)

The reactor building is not leak-tight; it has several ventilation systems, which — if still operating — will collect the air in the reactor building and exhaust it out the plant stack. A substantial release to the environment is possible regardless of whether the release is at the plant stack (100 meters above local grade), or at or near ground level due to structural failure or leakage of the reactor building. (The IAEA issue book for WWER-1000 NPPs does not address this issue /25/.)

The Czech Government report in March 2001 cites the Koeberg NPP in South Africa as having a similar arrangement /24/. That is as may be, but South Africa is not a member state of the European Union. No PWR NPP within the member states of the European Union has such a containment configuration.

The Czech report also states that the delay until the reactor cavity melt-through occurs provides time to implement emergency planning provisions /24/. Again, this may be, but emergency planning provisions will not prevent the contamination of Austrian (and Czech) territory in the event of a core melt accident where reactor cavity melt-through occurs. In addition, while the Czech report cites a value of 24 hours for the delay time, information presented at the 4 April 2001 expert meeting cited a value of 18 hours. In either case, it has to be noted that this is much faster than corresponding values for PWR NPPs in the member states of the European Union (generally in the range of 3-5 days). It also has to be noted that the resulting source term for Temelín will be much larger than the source terms for melt-through accidents at PWR NPPs in the member states of the European Union because those NPPs release the core debris into the ground rather than into a compartment above ground as is the case for Temelín. In addition, most (more than three-quarters) of the PWR NPPs in the member states of the European Union have
filtered venting systems which can be used to attenuate the source term before melt-through occurs; Temelín lacks such a system.

7.4.1.4 The Potential for Late Overpressure Failure of the Containment

Containment overpressurisation from steam and non-condensable gases is not expected to be a short-term threat to containment integrity owing to the large free volume and structural strength of the containment for quasi-static loads. However, in the longer term (3-5 days), if no containment heat removal occurs, the containment will eventually overpressurise (provided that no other mode of containment failure occurs first, such as liner failure due to DCH or hydrogen combustion, or such as reactor-cavity melt-through).

This late containment failure mode will result in some level of source term in addition to noble gases being released to the atmosphere. The magnitude of the release from other than noble gases depends on the location of the failure (i.e., whether the failure location is within the reactor building or above it) and dynamics after the failure (i.e., if the failure is within the reactor building, what is the response of the reactor building to the pressure transient and possible hydrogen combustion).

It is common practice within the European Union to use filtered vented containment systems to transfer such late overpressure failure into a controlled depressurisation of the containment. It was concluded in 1994 that late overpressurisation can be reasonably treated with filtered venting devices /17/. Filtered venting systems have been implemented at PWRs in Belgium, France, Germany, and the Netherlands, as well as in the non-EU western European nation of Switzerland /11/. In short, 78 out of 84 PWRs in western Europe (including the EU Member States and Switzerland) have implemented filtered venting system, thus filtered venting systems are state-of-the-art within the EU. (The IAEA issue book for WWER-1000 NPPs does not address this issue except in the context of information available to the operator to decide when to initiate venting /25/. The Czech report does not address late overpressure failure, except to state that it was studied in the context of developing severe accident management guidance /24/.)

7.4.2 Containment Bypass

A Level 2 probabilistic safety assessment (PSA) for Temelín was performed and completed in late 1995. In terms of Level 1 scope (core damage frequency), the PSA found that severe accidents initiated by steam generator tube rupture (SGTR) and steam generator collector leakage (SGCL) contribute about two-thirds of the CDF from all causes (including internal initiating events, fires, and floods) /18/. (The IAEA issue book for WWER-1000 NPPs does not address this issue in the context of its contribution to core damage frequency /25/.)

The Level 2 PSA results indicate clearly, as expected, that SGTR and SGCL severe accidents bypass the containment and are expected to be associated with large source terms (release fractions of cesium and iodine in excess of a few percent of core inventory). (This is consistent with the German Risk Study, which estimated cesium and iodine releases from SGTR sequences at 15% if the tube break location is not covered with water, and at 2.5% if the break location is covered by water /16/.)

The PSA is currently being updated, however no updated results were available at the 4 April 2001 expert meeting in Prague. For the moment, then, we are left with the 1995
PSA as being the best currently available indication of the risk profile of the Temelín plant.

In absolute terms, the combined frequency of SGTR and SGCL severe accidents is of the order of $6.6 \times 10^{-5}$ per year. These PSA results for containment bypass sequences due to primary-to-secondary leakage — both in absolute CDF terms and in terms of the dominance of containment bypass compared to other CDF contributors — are well above those for contemporary PSAs for PWR NPPs within the EU. The result is in excess of large release safety targets and limits which have been adopted for five of the seven EU member states with nuclear power plants (Finland, France, Germany, Sweden, and the United Kingdom) /21-23/, which range between $10^{-6}$ and $10^{-7}$ per year, and also well above the IAEA INSAG safety target for large releases for existing NPPs, which is less than $10^{-5}$ per year /2/.

It is important to recognise that SGTR initiating events occur at WWER-1000 reactors about ten times more often (on a per reactor-year basis) than at PWRs generally. Historically, SGTR initiating events have been caused by loose parts and primary water stress corrosion cracking (PWSCC). In order to limit the initiating event frequency of SGTR, it is therefore important that these factors be well controlled at Temelín.

The Temelín PSA found that human errors are the dominant source of core damage after SGTR and SGCL accidents, and this is not a surprising result since the automatic control response for such events in a WWER-1000 NPP is not designed for such events, and in the long term the automatic system response to contrary to what is required for such events /19/. Thus, a number of manual operator actions are required in order to bring SGTR and SGCL initiating events to a successful resolution and avoid core damage. The German Risk Study found similar results for SGTR sequences; following that study, automatic control systems were installed on German PWRs to response to SGTR sequences /20/.

The Czech Government's report to the trilateral process acknowledges that the PSA shows that primary to secondary leakage accidents are the most important contributor to total CDF and "large early release frequency (LERF). The Czech report makes the following points regarding these results /24/:

(a) The PSA had a design freeze date at the beginning of 1994, and the PSA does not reflect "a lot of safety improvements [which] were implemented into the design".

(b) The PSA results are preliminary and conservative because the Temelín steam generators are "of the latest design" and include improvements which reduce the likelihood of SG primary header cover failures and the header itself, thus reducing the relative CDF contribution of subsequent accident sequences.

(c) The generic steam generator cover leakage frequency in the PSA was "conservatively" based on an event in a WWER-440 plant, for which the majority of WWER-1000 PSAs has been excluded as not applicable to the WWER-1000 design. The IAEA IPERS mission recommended to make the frequency less conservative based on design differences and protective measures adopted in the Temelín design.
(d) The Temelín design measures "focused mainly" on primary to secondary leakage accidents with the following measures:

- SG primary header cover design is improved to limit the leakage flow rate significantly.
- Technical Specifications restrict plant operation even under very small primary to secondary leakage conditions.
- Special surveillance (eddy-current) inspections of SG header cover and header cover fixing bolts to prevent potential leakage.
- New symptom based emergency operating procedures address primary to secondary leakage sequences very carefully as those were identified dominating for the risk.
- Operator training at a full-scope simulator is focused on this risk dominating accident sequences.

Concerning these statements, the following remarks can be made as regards the PSA results:

(a) The PSA indeed reports a freeze date of September 1993, however the design analysed in the PSA includes certain planned modifications which were credited in the analysis. These credited modifications include those related to the SG collector restraining bolts and implementation of symptom-oriented EOPs. These modifications are fully reflected in the PSA, and rather than making the PSA conservative (as the Czech report asserts the results are), they make the results more realistic.

(b) Failures of the SG header itself were excluded from the PSA based on design changes. Thus, the PSA shows no CDF contribution from SG collector failures. The contribution to CDF comes from header leakage (not failure), and fully reflects the design modification to limit flow to 40 mm effective leak rate (as opposed to the 100 mm leak rate in the original design). Again, this is already reflected in the PSA, rendering the results more realistic (instead of conservative as asserted by the Czech report).

(c) The SG collector leakage frequency is recognised as conservative by the Austrian side. Based on additional operating experience (without a subsequent failure) in WWER NPPs since the end of data collection for the PSA, we estimate that the SGCL initiating event frequency should be reduced by a factor of about 2 to $1.3 \times 10^{-4}$ per year. This would reduce the SGCL CDF value from $4.33 \times 10^{-5}$ per year to $2.17 \times 10^{-5}$ per year.

(d) Similarly, the SGTR initiating frequency can be updated based on operating experience since the end of PSA data collection. This would result in a reduction of this initiating event frequency also by a factor of about 2 to $2.1 \times 10^{-2}$ per year. This would reduce the SGTR CDF from $2.27 \times 10^{-5}$ per year to $1.08 \times 10^{-5}$ per year.

(e) One other correction should be made to the PSA results, namely the deletion of a large LOCA frequency (involving failure only of accumulators, but with successful low pressure injection). This is not, as acknowledged by the Czech experts at the Prague meeting on 4 April 2001, a core damage sequence. This would reduce the CDF contribution from non-bypass sequences from $2.36 \times 10^{-5}$ per year to $1.99 \times 10^{-5}$ per year.
year. With these three corrections, the internal events CDF would drop from $8.96 \times 10^{-5}$ per year to $5.24 \times 10^{-5}$ per year. Containment bypass sequences (SGCL plus SGTR) would still contribute $3.25 \times 10^{-5}$ per year, or about 62% of the revised CDF. Even with the corrections, containment bypass sequences still dominate CDF and still contribute at a level above the INSAG safety target for large release frequency for existing NPPs of $1 \times 10^{-5}$ per year /2/.

(f) As regards the specific measures cited by the Czech report (cited in item "d" above in the discussion of the Czech report):

— The SG primary header cover design improvement is already included in the PSA results.

— Restriction of plant operation under small primary to secondary leakage rates is standard practice in the member states of the European Union. However, such restrictions have not historically prevented SGTR events in western NPPs, and to the extent that they do reduce the likelihood of such events this is already reflected in the frequency of occurrence.

— Regarding eddy-current testing of the SG header cover and header cover fixing bolts, it has to be noted that SG tubes are also eddy-current tested, and SGTRs nonetheless continue to occur in western PWRs. This measure helps, but it is not a cure-all and there is insufficient data in the specific application to SG header cover and SG header cover fixing bolts to justify changing the frequency SGCL events.

— The symptom based EOPs are already included in the PSA results. Indeed, the PSA included a sensitivity study, which shows that even if the rate of all human errors is reduced by factors of 2, 5, and 20, the internal events CDF changes to values of $5 \times 10^{-5}$, $3 \times 10^{-5}$, and $2 \times 10^{-5}$ per year, respectively. A 20-fold reduction in human error rates over those claimed in the PSA cannot credibly be considered, and lesser reductions still leave a CDF above $1 \times 10^{-5}$ per year with still significant contributions from containment bypass sequences.

— Operator training at a full-scope simulator based on containment bypass sequences is commendable, but it has to be noted that the human error rates in the PSA fully reflected these considerations.

The PSA update was started in January 2001, and is expected to take 15 months /26/. Thus, we cannot reasonably expect revised PSA results until about March 2002, which is well beyond the horizon of the Melk protocol. Accordingly, and in the absence of better information, we regard the PSA results (modified as above) as indicative of the nature of the problem. Containment bypass events are thus considered to be major contributors to CDF, and at a frequency well above those for PWR NPPs in member states of the European Union.

7.5 References


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