Austrian Technical Position Paper

Safety Aspects of Temelín Nuclear Power Plant

July 2001

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Disclaimer

The international team of experts commissioned by the Austrian government selected 29 issues of concern based on their expert judgement and documents available to them as of January 2001. Although the intent of the experts was to focus on the most important issues, there is no assurance that all important issues have been addressed nor that these 29 were fully covered due to the limited documentation available for review and the limited time available to assess that documentation. Individually, the experts take responsibility for those parts of the report that lie within their competence.

The experts were not tasked with performing a licensing review of Temelín, and nothing in this report may be construed to represent any such review. The responsibility for the safety and licensing of Temelín rests with CEZ a.s. as the owner of the facility, and with the SÚJB, as the designated nuclear regulatory authority under Czech law.

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Abstract

Review of documentation and discussions with Czech experts revealed that important safety issues were not resolved and in some essential cases the relevant analyses were not performed or completed. The most important unresolved issues are considered to be:

- reactor pressure vessel integrity – no pre-service pressurised thermal shock analysis corresponding to European state-of-the-art was performed,
- pre-service non-destructive testing – the testing procedure is not yet validated and no state-of-the-art Tandem or French focussing technique was used,
- environmental and seismic qualification of safety relevant equipment – the qualification procedure is not yet completed,
- site seismicity – no state-of-the-art methods for dating of the near-by tectonic faults were applied,
- high energy pipelines at the +28.8 m level – the lines running in parallel are not sufficiently separated so as to preclude multiple dependent failures in case of line rupture,
- qualification of valves for steam-water mixtures – the functional qualification is not yet completed,
- containment integrity under accident conditions – several accident scenarios involving containment leakage due to hydrogen detonations and melt/wall contact or reactor cavity melt-through have not been sufficiently analysed in view of a containment design that is unique within EU PWRs and it has not been demonstrated that the dominance of containment by-pass sequences as a contributor to core damage frequency, which is by far not in line with the European consensus, can be significantly reduced.

Neither for Temelin Unit 1 nor Unit 2 would European state-of-the-art practice permit operation or even fuel loading before resolution of issues as addressed above, concerning in particular reactor pressure vessel, non-destructive testing, equipment qualification, +28.8 m level piping or containment behaviour under core melt accident conditions.

The fact that start-up of Unit 1 was begun in spite of these deficiencies raises questions regarding the attitude of the operator and of the licensing authority toward the requirements of safety culture.

Thus nuclear safety of Temelín NPP cannot yet be considered proven to be in line with the state-of-the-art in member states of the European Union. The essential analyses could be accomplished within one year; their results must be known in order to estimate the (possibly considerable) resources and time required for corrective measures to resolve the most serious of these issues regarding embrittlement of the reactor pressure vessel, pressure line routing at the +28.8 m level, containment integrity, and seismic safety. The implementation of the requested analyses is therefore of primary importance.

Shrnutí

Prostudování dokumentace a diskuse s českými experty ukázaly, že důležité bezpečnostní otázky nebyly vyřešeny a zě se v některých závažných případech neprovedla nutná analýza nebo není ukončená. Za velmi důležité otázky považujeme:

- integritu reaktorové nádoby – neprovedly se předprovozní analýzy odolnosti vůči křehnutí podle dnešního stavu techniky,
- nedestruktivní testy – předprovozní zkoušky, které nebyly kalibrovány a provedeny s odpovídající ultrazvukovou testovací metodou podle stavu techniky s Tandem technique nebo francouzskou focussing technique,
- kvalifikaci bezpečnostně významných komponent za podmínek při nehodách a zemětřesení- kvalifikační program ještě nebyl dokončen,
- seismické ohrožení lokality - nepoužila se moderní metoda podle stavu techniky k datování tektonických zlomů probíhajících blízko lokality,
- vysokoenergetické potrubí na podlaží +28.8 m - probíhají souběžně a nejsou dostatečně chráněny k zabránění několikanásobnému selhání při roztržení potrubí,
- kvalifikaci ventilů pro parovodní směs - funkční kvalifikace doposud nebyla provedena,
- integritu kontejnmentu během havárijních podmínek – některé havarijní scénáře sehlhání kontejnmentu způsobené explosí vodíku, kontakt tavenina/stěna nebo protavení šachty reaktoru nebyly dostatečně analyzovány ohledně toho, že se jedná o unikátní konstrukci kontenjnmentu pro PWR v členských státech EU. Dále se nepovedlo dokázat, že by bylo možné potlačit dominántní příspěvek událostí s obtokem kontejnmentu k protavení aktivní zóny (Core Damage Frequency), který je v rozporu s evropským bezpečnostním konsensem.

Podle Evropské praxe licensování by se nepovolil provoz pro 1. blok ani pro 2. blok, ani zavezení paliva v Jaderné elektrárny Temelín před úplným vyřešením výše zmíněných bezpečnostních otázek, především integrity reaktorové nádoby, zkoušky jakosti materiálu, kvalifikace zařízení, podlaží +28.8 m a chování kontejmentu při haváriích s tavením aktivní zóny.

Skutečnost, že se najíždí 1. blok přes chybějící analýzy a opatření, vyvolává obavy ohledně přístupu provozovatele a dozoru vůči požadavku bezpečnostní kultury.

Z těchto důvodů se doposud neprokázalo, že jaderná bezpečnost JETE Temelín odpovídá stavu techniky, který se používá ve státech EU. Nutné analýzy sice lze provést během jednoho roku; k odhadnutí (možná velké) materiálové a časové náročnosti nápravných opatření k řešení závažných bezpečnostních otázek, jako jsou křehnutí reaktorové nádoby, vedení vysokoenergetických potrubí na podlaží +28.8 m a integrita kontejnmentu, je provedení již zmíněných analýz předpokladem. Provedení analýz proto má nejvyšší význam.

Kurzfassung

Sichtung der Dokumentation und Diskussion mit tschechischen Experten ergaben, dass wichtige Sicherheitsfragen nicht gelöst wurden und dass in manchen schwerwiegenden Fällen auch die notwendigen Analysen noch nicht gemacht oder nicht abgeschlossen sind. Als besonders wichtige Problembereiche werden betrachtet:

- Integrität des Reaktordruckbehälters – es wurde keine dem heutigen europäischen Stand der Technik entsprechende vorbetriebliche Sprödbruchsicherheitsanalyse durchgeführt,
- Zerstörungsfreie Prüfung – die vorbetrieblich angewandten Verfahren wurden nicht kalibriert und es wurde keine dem Stand der Technik entsprechende Ultraschallprüfung mit Tandem-Verfahren oder französischem Verfahren mit fokussierenden Prüfköpfen eingesetzt,
- Qualifikation sicherheitsrelevanter Komponenten unter Störfallbedingung und Erdbebenbelastung – das Qualifikationsverfahren ist nicht abgeschlossen,
- Erdbebengefährdung des Standortes – es wurden keine dem Stand der Technik entsprechenden Verfahren zur Datierung der standortnahen tektonischen Störungen eingesetzt,
- Hochbeanspruchte Rohrleitungen auf der +28,8 m Bühne – parallel geführte Leitungen sind unzureichend gegeneinander abgeschirmt um bei Leitungsbruch mehrfaches Folgeversagen zu verhindern,
- Qualifikation von Ventilen für Dampf-Wassergemisch – die funktionelle Qualifikation liegt bisher nicht vor,
- Containmentintegrität unter Unfallbedingungen – einige Unfallszenarien, mit Containmentversagen durch Wasserstoffexplosionen, Schmelze/Wand-Kontakten oder Durchschmelzen der Reaktorgrube wurden im Hinblick auf die in den europäischen Mitgliedstaaten bei DWR beispiellose Containmentbauweise nicht hinreichend analysiert, außerdem konnte bislang nicht gezeigt werden, daß die dem europäischen Sicherheitskonsens zuwiderlaufende Dominanz des Beitrages von Containmentbypass-Ereignissen zur Kernschmelzwahrscheinlichkeit unterdrückt werden kann.

Nach europäischer Genehmigungspraxis dürften weder Temelin Block 1 noch Block 2 betrieben oder auch nur mit Brennstoff beladen werden, bevor oben genannte Unsicherheiten insbesondere betreffend Reaktordruckbehälter, Werkstoffprüfung, Komponentenqualifizierung, +28,8 m-Bühne und Containmentverhalten bei Kernschmelzstörfällen beseitigt sind.

Die Tatsache, dass Block 1 trotz der fehlenden Analysen und Maßnahmen hochgefahren wird, lässt Fragen hinsichtlich der Einstellung des Betreibers und der Aufsichtbehörde zu den Erfordernissen der Nuklearen Sicherheitskultur aufkommen.

Es kann somit bislang nicht als nachgewiesen betrachtet werden, daß die nukleare Sicherheit des KKW Temelín dem Stand der Technik, wie er in den Mitgliedstaaten der Europäischen Union zur Anwendung kommt, entspricht. Die erforderlichen Analysen könnten zwar innerhalb eines Jahres durchgeführt werden; um allerdings zu den Schlüsselfragen Reaktordruckbehälterversprödung, Leitungsführung auf der +28,8 m-Bühne, Containmentintegrität und Erdbebensicherheit den (möglicherweise erheblichen) materiellen und zeitlichen Aufwand an Nachrüstmaßnahmen abschätzen zu können, ist das Vorliegen der Ergebnisse der angesprochenen Analysen unabdingbar. Der Durchführung der Analysen kommt somit vorrangige Bedeutung zu.

Executive Summary

With the ”Protocol of the negotiations between the Czech and the Austrian Government led by Prime Minister Zeman and Federal Chancellor Schüssel with the participation of Commissioner Verheugen” of 12 December 2000 (the Melk Protocol) both sides agreed with Commissioner Verheugen to conduct a ”trialogue” to find a better mutual understanding on the issue of the Temelin Nuclear Power Plant. To this end, the Governments of Austria and the Czech Republic requested the Commission to undertake an expert mission with trilateral participation. The aim of this mission was to facilitate the dialogue between the Governments of Austria and the Czech Republic on the issue of nuclear safety.

The present report was written close to the end of this ”trialogue”. It represents the position of the international team of experts commissioned by Austria, based on the information that was made available to the team at the outset of the ”trialogue” and before regarding 29 issues of concern that were presented by Austrian experts. This information was evaluated and assessed ”on the basis of the state-of-the-art relevant in the Member States of the European Union”, as referred to in the ”Melk Protocol”.

Some issues could be regarded as closed - in view of the information received - for the purpose of the Melk process. Others have been deferred to further bilateral discussion or to the ongoing Environmental Impact Assessment (Chapter V of the Melk Protocol). But for a number of issues there is convincing evidence of clear and significant deviations from European state-of-the-art practice, aggravated by the fact that these issues have not been resolved before nuclear start up of Unit 1. Although some of these problems have been partially addressed, or announced to be addressed in the future respectively, by the Czech side, there remains a clear need of urgent corrective actions.

For the Austrian side, the following are the most important concerns related to the open issues:

1. <u>No Pressurised Thermal Shock (PTS) analysis done for Unit 1 before start-up</u>

No pre-service structural integrity assessment for pressurised thermal shock (PTS) conditions was performed for the reactor pressure vessel (RPV) of Unit°1 before nuclear operation. The simplified calculations of operational limit curves provided as substitute are not appropriate to prove structural integrity as required by all applicable codes.

The integrity of the RPV is of utmost importance as safety systems in Temelin - as elsewhere - are not designed to compensate rupture of the RPV. Such an accident could result in release of radioactive materials into the environment that could have direct impacts on Austria. A pre-operational PTS analysis serves to prove sufficient safety margin against PTS induced catastrophic failure of the RPV throughout its entire service life and can have considerable influence on the licensing process of a plant. Because of the high nickel content of the RPV steel Temelin might not meet the PTS resistance design safety criterion under the present operating conditions. Early incisive measures, such as core reconfiguration may be necessary. Speculating on thermal annealing of the RPV – a disputed measure to prolong service life – already before first operation is basically incompatible with European safety standards. The fact that the licensing authority has agreed to a schedule that requires a complete PTS analysis only within the next 5 years - and thus evades an important licensing prerogative raises serious questions about the attitude of parties involved towards safety culture.

2. Non Destructive Testing programme not state-of-the-art

The non-destructive testing (NDT) procedure for primary loop components involving a variety of methods has not yet been qualified (calibrated) although appropriate test blocks are available. Material defects undetected by insufficient NDT could jeopardise the strength and thus component integrity.

No dedicated ultrasonic test methods (i.e. Tandem technique or French focussing technique) were used that guarantee a reliable detection of severe crack-like defects perpendicular to the surface in the reactor pressure vessel wall.

In the plant documentation three non-allowable defect indications (according to the accepted standards) were discovered, that were left unrepaired without the appropriate defect evaluations.

In spite of start-up of Unit 1 the final documents of the ultrasonic tests that were performed on the primary loop are not available at the plant. There is no summary document on all pre-service NDT results as required by the licensing body's guidelines.

The circumferential welds of the main steam lines on the +28.8 m level were tested using X-ray radiography only. The wedges for the whip restraints welded onto the main steam lines were not inspected after welding (only an acceptance test weld was inspected), in-service inspection (ISI) was performed on one of 16 fixations only.

3. Environmental and Seismic Equipment Qualification not completed

Qualification of equipment important for safety is not fully established even though the plant is already in the nuclear commissioning phase. This is not in agreement with internationally accepted safety requirements, by which proof of the qualification of equipment important for safety has to be established and demonstrated before fuel loading. Deviations are to be treated as exceptions requiring approval according to special rules of the licensing body.

The formal qualification reports, which should have been issued, reviewed, approved and available on site before fuel loading, are not yet fully issued. No information has been made available about the justifications brought forward by the Operator to receive authorisation for fuel loading without having fully established Equipment Qualification nor about conditions imposed by the Regulator.

In view of the qualification programme in progress at Temelin, the programme needs to be further explained in order to: (a) ascertain the date of expected completion of qualification of safety and safety related equipment; (b) understand the licensing aspects of the delayed qualification of equipment, including management and documentation; and (c) to obtain information on the current technical outcomes for specific equipment.

4. Seismic hazard assessment not state-of-the-art

Seismic Hazard Assessment for Temelin NPP is apparently not based on state-of-the-art methods. There are indications that the currently assumed Safe Shut Down Earthquake SSE of 0,1 g maximum horizontal peak ground acceleration might underestimate possible earthquake loads and the related risk.

Two major tectonic faults (Jachymov/Hluboka and Blanice) pass Temelin NPP site within 5 and 13 km distance, some smaller ones are even closer. It is claimed that these faults are inactive but for proof no state-of-the-art dating methodology has been applied.

5. Design weakness at the +28.8 m elevation level

Main steam and feedwater lines at the +28.8 m elevation level run in parallel between the penetration of the containment and the main isolation valves over a distance of some 20 meters. Physical separation (e. g. concrete walls) from each other and from other safety relevant equipment has not been installed.

In case of a rupture of one or more of these lines damage of adjacent lines as well as other safety-relevant equipment cannot be excluded as a consequence of pipe whip and/or jet impingement effects by discharged material. This could trigger an accident sequence causing large radioactive releases. This issue has not been sufficiently addressed.

Comprehensive analyses of multiple line ruptures are necessary to make valid estimates of the severity of the consequences of such an event. The analyses must provide detailed information on reactivity control during such accidents, the subcooling effects on the pressure vessel and steam generators and the related threat of pressurised thermal shocks. Issuance of a complete safety case encompassing all aspects related to this issue is essential. This should include among others, identification of all potential internal and external initiators, water-hammer and dynamic effects in feedwater and steam lines as well as identification of limiting conditions related to acceptable break sizes. Furthermore this safety case has to include envisaged reconstruction or re-positioning/re-routing of safety-important components/equipment and of the steam and feedwater lines. Robustness and adequacy of installed pipe whip restraints installed at containment penetration and at partition wall towards the machine hall must be included as well as proof of erosion-corrosion prevention and mitigation.

The main objective of adequate re-assessment and reconstruction of the +28.8 m level must be to physically exclude multiple steam line breaks and consequential component and equipment failures that cannot be compensated by the safety systems and thus could result in severe accidents with potential large release of radioactivity.

6. Functional qualification of safety related valves open

For the main steam relief and safety valves the functional qualification is still pending. Non qualified valves could remain stuck open in case of accident operation under two phase flow conditions. This could trigger an event sequence resulting in a severe accident with large release of radioactivity. In addition, isolation valves on the main steam lines upstream of the relief valves, which could mitigate the adverse consequences of a stuck open valve, are not installed in Temelin.

7. Containment integrity not ensured

Several problem areas, which could endanger the integrity of the containment in case of severe accidents, are considered to be not addressed adequately by the Czech side:

Hydrogen problem: A system of passive autocatalytic recombiners (PAR) for hydrogen generated during severe accidents has been installed, but size and placement are not adjusted to severe accident conditions. Additional state-of-the-art analyses and upgrade of the PAR system is required.

High Pressure Core Melt Ejection: High pressure ejection of core melt debris from the reactor vessel in case of severe accident can result in high containment leakage. A depressurisation system needs to be installed and analyses of the mechanical and thermal consequences in such accident sequences need to be performed.

Reactor Cavity Melt-Through: Unimpeded interaction between molten core debris and reactor cavity can result in melt-through of the reactor cavity and release of core debris and airborne radioactive materials via the reactor building to the environment. Analyses of possible modifications to the reactor cavity door and of means to cool reactor debris are required.

Late Overpressure Failure of Containment: Continued pressurisation of the containment during a severe accident sequence can result in overpressure failure of the containment. Installation of a filtered venting system for the containment might help to resolve this issue.

Containment bypass accidents: Two types of accidents leading to containment bypass (steam generator tube rupture and steam generator collector leakage) contribute about two thirds of the frequency of core damage, i.e. to possibly severe accidents. European practice requires that bypass accidents not be dominant contributors to CDF.

8. Severe Accident Management Guidelines not in place

The implementation of Severe Accident Management Guidelines (SAMGs) as a prerequisite for adequate Defence in Depth provisions is accepted western safety standard. At Temelin NPP, such guidelines have not yet been developed and implemented, although CEZ confirms that they are a necessity, and although they would be of special importance for Temelin, being a first case of significant composite technology WWER-1000 plant.

9. Safety Culture

Appropriate plant design, construction and operation on the one hand, the financial stability, competence and commitment of the Operator and the Regulator to nuclear safety and the strength and independence of the latter on the other are the preconditions for the necessary high level of safety in the operation of a nuclear facility.

The safety issues dealt with in the trilateral meetings and workshops under the Melk process were originally mainly concerned with matters of plant design and assessment: issues of technical nature. During the presentations and discussions, however, situations arose, which raised concern regarding commitment to nuclear safety and responsibility of Operator and Regulator, their roles as well as the capability of the Regulator to take independent decisions. Thus the issue of safety culture related to the licensing process gained high priority.

Commissioning and licensing have progressed in spite of the lack of important elements of such a process: non destructive testing of important primary loop components is incomplete, preservice pressurised thermal shock analysis is lacking, inadequate evaluation of containment behaviour in severe accident conditions, analysis and resolution of the high energy line break issue at the +28.8 m level is only marginal, the environmental and seismic equipment qualification is not fully established and functional qualification of important safety relevant components is still open, almost no consideration to PSA study performed by CEZ as support tool for many decisions in the safety assessment, severe accident management guidelines still to be developed. There is concern how defense-in-depth can be maintained, when important issues concerning RPV, containment and SAMGs are left unresolved in the licensing process. The Austrian side is also alarmed by the fact that during the discussion of safety issues and related licensing aspects the Regulator in no instance showed his role as distinct from that of the Operator and there was no discernible willingness on the part of the official representative of the Regulator to openly evaluate important safety concerns or suggestions put forward – neither those addressed to the Operator, nor those addressed to his own organisation.

Neither for Temelin Unit 1 nor Unit 2 would European state-of-the-art practice permit operation or even fuel loading before resolution of issues as addressed above, concerning in particular reactor pressure vessel integrity, pre-service non-destructive testing, environmental and seismic equipment qualification, physical separation of high energy lines at the +28.8 m level, functional qualification of valves, or containment behaviour under core melt accident conditions.

Thus nuclear safety of Temelín NPP cannot yet be considered proven to be in line with the state-of-the-art in Member States of the European Union. The essential analyses could be accomplished within one year; their results must be known in order to estimate the (possibly considerable) resources and time required for corrective measures to resolve the most serious of these issues regarding embrittlement of the reactor pressure vessel, pressure line routing at the +28.8 m level, containment integrity, and seismic safety. The implementation of the requested analyses is therefore of primary importance.

Souhrn

V „Protokolu z jednání mezi českou a rakouskou vládou, vedených mezi předsedou vlády Zemanem a spolkovým kancléřem Schusselem za účasti komisaře Verheugena“ z 12.12.2000 (Protokol z Melku), se obě strany shodují s komisařem Verheugenem na vedení "trialogu" k dosažení lepšího vzájemného porozumění v otázce Jaderné elektrárny Temelín.Z tohoto důvodu se shodly vlády Rakouska a ČR s Evropskou komisí na provedení expertní mise s třístrannou účastí. Cílem této mise je usnadnit dialog o otázkách jaderné bezpečnosti mezi vládami Rakouska a České republiky.

Předložená zpráva byla zpracována krátce před koncem této třístranné diskuse. Zpráva reprezentuje stanovisko mezinárodních expertů kteří byly pověřeni Rakouskem, a vychází z informací poskytnutých na začátku „trialogu“ a předtím k 29 otázkám, které budily obavy a byly prezentovány rakouskými experty. Tato informace byla vyhodnocena a posouzena v souladu s Protokolem z Melku „na základě nejnovějších postupů, uplatňovaných v členských státech Evropské unie“.

Některé bezpečnostní otázky lze – vzhledem k poskytnuté informaci – označit jako ukončené v rámci Protokolu z Melku. Další se posunuly k projednání na bilaterální úroveň nebo do EIA (Protokol z Melku). Pro celou řadu otázek existují přesvědčivé důkazy pro jasné a závažné odchylky od evropské praxe a evropských standardů. To bylo zhoršeno skutečností, že tyto problémy nebyly rozřešeny před jaderným spuštěním prvního bloku. Ačkoliv česká strana několik těchto problémů částečně řešila, resp. oznámila jejich řešení v budoucnosti, zůstala i nadále nutnost urgentního zlepšení.

Následující problémy jsou těmi nejdůležitějšími pro rakouskou stranu v souvislosti s otevřenými otázkami:

1. <u>Neprovedení analýzy tepelného šoku (PTS) pro 1. blok před spouštěním reaktoru</u>

Pro 1. blok nebyla provedena předprovozní analýza odolnosti reaktorové nádoby vůči tepelnému šoku pod tlakem. Zjednodušené výpočty provozních limitních křivek, které byly poskytnuty jako náhražka, nejsou vhodné k prokázání strukturální integrity, kterou vyžaduje každý vhodný výpočtový program (kód).

Integrita reaktorové nádoby má nejvyšší význam, protože bezpečnostní systémy v Temelíně - ani jinde - nejsou projektované tak, aby zvládaly roztržení reaktorové nádoby. Taková havárie by mohla vést k úniku radioaktivních látek do okolí, který může mít přímé následky v Rakousku. Předprovozní analýza tepelného šoku slouží k prokázání, že během celé životnosti se zachovává dostatečně velká bezpečnostní rezerva vůči katastrofálnímu selhání reaktorové nádoby vyvolané tepelným šokem a může mít značný vliv na licencování elektrárny. Protože ocel nádoby obsahuje mnoho niklu, Temelín možná nesplňuje toto bezpečnostní kritérium – design s dostatečnou odolností vůči PTS za aktuálních provozních podmínek. Včasná zásadní opatření, jako rekonfigurace aktivní zóny, by mohla být nutná. Spekulovat s provedením žíhání nádoby – sporné opatření s cílem prodloužení životnosti – již před prvním provozem je zásadně v rozporu s evropskými bezpečnostními standardy. Skutečnost, že povolovací úřad schválil harmonogram, který si vyžaduje úplnou analýzu tepelného šoku teprve v průběhu příštích pěti let a tím vynechává důležitou podmínku pro licencování, budí vážné otázky o postoji stran, které se zabývají bezpečnostní kulturou.

2. Program pro nedestruktivní testy (NDT) neodpovídá stavu techniky

Program pro nedestruktivní testy komponent primárního okruhu ještě nebyl kvalifikován (kalibrován), ačkoliv k tomu potřebné testovací bloky jsou k dispozici. Nedostatky materiálu, které by zůstaly nezjištěné díky nedostačujícím NDT, by mohly ohrozit pevnost a tím také integritu komponent.

Nepoužily se specializované ultrazvukové testovací metody (v tomto případě Tandem technique nebo francouzská focussing technique), které by zaručily spolehlivé zjištění větších trhlinových defektů, kolmých k povrchu zdi reaktorové nádoby.

V dokumentaci komponent byly objeveny poukazy na tři podle norem nepřípustné nedostatky, které byly ponechány bez předepsaného vyhodnocení chyby.

Ačkoliv se zařízení již spustilo, elektrárna nemá k dispozici zprávu o všech ultrazvukových testech primárního okruhu. Neexistuje ani souhrnná zpráva o všech výsledcích předprovozních NPT, jak to požadují pravidla jaderného dozoru.

Obvodové svary na parovodu na podlaží +28,8m byly kontrolovány jenom pomocí rentgenového záření radiografií. Klíny pro omezovače švihu, které byly navařovány na parovody, se po navaření nekontrolovaly (zkoušel se jenom svařovací vzorek), standardní inspekce po uvedení do provozu (in-service inspection) se provedla jenom u jednoho z 16 omezovačů.

3. Nedokončená kvalifikace komponent pro podmínky za nehod

Kvalifikace bezpečnostně významných komponent nebyla úplně provedena pro neobvyklé stavy, které mohou nastat při nehodách a zemětřesení, ačkoliv se elektrárna již nachází ve stavu jaderného zkušebního provozu. To je v rozporu s mezinárodně akceptovanými bezpečnostními požadavky, které vycházejí z toho, že se důkaz o kvalifikaci komponent významných pro bezpečnost provede a doloží každopádně před zavezením paliva. S odchylky se musí zacházet jako s výjimky, které se vyžadují schválení v souladu se speciálními pravidly jaderného dozoru.

Formální kvalifikační zprávy, které by měly být zpracovány, hodnoceny a schváleny a být v elektrárně k dispozici před zavezením paliva, ještě nejsou zcela vydané. Žádné informace nejsou k dispozici o tom, jakého ospravedlnění použil provozovatel k tomu, aby obdržel povolení k zavezení paliva bez úplného dokončení kvalifikace komponent, ani o tom, které podmínky uložil dozoru.

Ke kvalifikačnímu programu, který v Temelíně probíhá, jsou nutná další vysvětlení pro (a) zjištění termínu, kdy se očekává dokončení kvalifikace bezpečnostního zařízení a zařízení významného pro bezpečnost (b) porozumnění licenčních aspektů zpožděné kvalifikace zařízení, včetně manažmentu a dokumentace a (c) obdržení informace o současném technickém výsledku pro specifická zařízení.

4. Hodnocení seismického rizika neodpovídá stavu techniky

Hodnocení seismického rizika podle všeho neodpovídá metodám nejnovějšího stavu techniky. Existují indikace, že aktuálně předpokládané Safe Shut Down Earthquake (zemětřesení s bezpečným odstavením reaktoru) o hodnotě 0,1 g (maximální zrychlení v úrovni volného terénu) by mohlo podhodnotit možnou śilu zemětřesení a s tím spojené skutečné riziko.

Dva větší tektonické zlomy (Jáchymov/Hluboká a Blanice) probíhají kolem elektrárny Temelín ve vzdálenosti 5 a 13 km, některé menší poruchy jsou ještě blíže. Tvrdí se, že tyto zlomy nejsou aktivní, nepoužila se však však žádná moderní metoda určení stáří k prokázání tohoto tvrzení.

5. Nedostatky konstrukce na podlaží +28,8 m

Potrubí ostré páry a napájecí vody probíhají souběžně na podlaží +28,8 m mezi kontejnmentem a hlavním uzavíracím ventilem přes ca. 20 m. Fyzikální separace (například betonové zdi) od sebe nebo od jiného bezpečnostního zařízení se neinstalovala. V případě roztržení jednoho nebo více takových potrubí nelze vyloučit poškození přilehlých potrubí nebo také jiného bezpečnostního zařízení, způsobené jako důsledek švihu potrubí anebo vystřelení ulomeného materiálu. To by mohlo iniciovat sekvenci havárií, která by zapříčinila velký únik radioaktivních látek. Tato otázka nebyla dostatečně řešena.

Obsáhlé analýzy mnohonásobného roztržení potrubí jsou nutné k obdržení věrohodných odhadů toho, jak vážné následky taková událost může mít. Analýzy musí obsahovat podrobné informace o kontrole reaktivity během takových haváriích, o efektu podchlazení na reaktorovou nádobu a parogenerátory a o souvisejícím nebezpečí tepelného šoku pod tlakem. Vypracování úplného maximálního scénáře (safety case), který obsahuje všechna hlediska v souvislosti s touto otázkou má rozhodující význam. To by mělo obsahovat mezi dalším: identifikaci všech potenciálních vnitřních a vnějších iniciátorů, včetně vodního rázu a dynamických efektů v parovodu a potrubí napájecí vody a identifikaci přípustných, limitních podmínek ve vztahu k přijatelné velikosti lomů. Dále by tento maximální scénář měl obsahovat oznámení rekonstrukce nebo nového uspořádání bezpečnostně významných komponent/zařízení a parovodu i potrubí napájecí vody. Pevnost a vhodnost instalovaných omezovačů švihu, namontovaných na průchodu kontejnmentem a na mezistěnu k strojovně, se musí zahrnout stejně jako důkaz o prevenci a zamezení eroze a koroze.

Nejdůležitějším cílem odpovídajícího opětného hodnocení a rekonstrukce na podlaží +28,8 m musí být fyzikální vyloučení mnohonásobného roztržení potrubí a následného selhání komponent a zařízení, které nemůže být nahrazeno bezpečnostními systémy, a proto by potenciálně mohlo vyvolat těžké havárie s velkým únikem radioaktivity.

6. Funkční kvalifikace bezpečnostně významných ventilů otevřená

Funkční kvalifikace přepouštěcích a pojistných ventilů stále není hotová. Nekvalifikované ventily by v případě nehody s prouděním parovodní směsi mohly být blokovány v otevřeném stavu. Tím se můžou iniciovat sekvence, které vedou k těžké havárií s velkým uvolněním radioaktivních látek. Dále v Temelíně nejsou instalovány na parovodu uzavírací ventily ve směru proudu nad přepouštěcími ventily, které by mohly zmírnit negativní následky zablokovaných otevřených ventilů.

7. Integrita kontejnmentu není zajištěná

Několik vážných problémů, které by mohly ohrozit integritu kontejnmentu v případě vzniku havárie, považujeme za nedostatečně vyřešené českou stranou:

Problém vodíku: Systém pasivních autokatalytických rekombinátorů (PAR) se instaloval pro vodík, vznikající během těžkých havárií, velikost a posice však neodpovídají podmínkám během těžkých havárií. Je nutné provést dodatečné analýzy a úpravy systému PAR podle nejnovějších metod.

Vysokotlaké vystřelení roztavené aktivní zóny: Vysokotlaké vystřelení částí taveniny z reaktorové nádoby mohou při těžkých haváriích způsobit velké úniky z kontejnmentu. Je nutné instalovat systém k potlačení tlaku a provést analýzy mechanických a termických následků takových havarijních sekvencí.

Protavení šachty reaktoru: Neomezená interakce mezi roztaveným materiálem aktivní zóny a šachtou reaktoru může vést k protavení šachty reaktoru a pronikání úlomků aktivní zóny a vzduchem neseného radioaktivního materiálu přes reaktorovnu do okolí. Je nutné provést analýzy možných modifikací u dveří šachty reaktoru a možností chlazení úlomků.

Zpožděné přetlakové selhání kontejnmentu: Pokračující přetlakování kontejnmentu během havarijních sekvencí může vyvolat přetlakové selhání kontejnmentu. K vyřešení tohoto problému může přispět instalace filtroventilačního systému pro kontejnment.

Havárie s obtokem kontejnmentu: Dva druhy havárií vedoucí k obtoku kontejnmentu (roztržení trubek parogenerátoru a únik z kolektoru parogenerátoru) přispívají asi ke dvěma třetinám z četnosti poruch aktivní zóny, to znamená, k možná těžkým nehodám. Evropská praxe požaduje, aby havárie s obtokem kontejnmentu nebyly dominantním příspěvkem k CDF (Core Damage Frequency).

8. Severe Accident Management Guidelines nejsou k dispozici

Zavedení Severe Accident Management Guidelines (SAMG) jako podmínky pro odpovídající ochranu do hloubky jsou akceptovaným západním bezpečnostním standardem. V elektrárně Temelín taková pravidla ještě nebyla vyvinuta a realizována, ačkoliv ČEZ potvrdil, že jsou nutné, a to zvlášť v případě Temelína jako prvního případu "kompozitní technologie" pro VVER-1000 elektrárny.

9. Bezpečnostní kultura

Vyhovující konstrukce elektrárny a výstavba a provoz elektrárny na jedné straně, finanční zajištění, kompetence a angažmá provozovatele a státního dohledu nad jadernou bezpečností, především jeho moc a nezávislost na straně druhé jsou podmínkou pro požádanou vysokou bezpečnostní úroveň při provozu jaderného zařízení.
Bezpečnostní otázky projednané během třístranných schůzek a workshopů v rámci Protokolu z Melku se ze začátku týkaly hlavně konstrukce elektrárny a ohodnocení

elektrárny: t.j. otázek technického rázu. Během prezentací a diskuzí však vznikly situace, které vyvolaly obavy ohledně angažmá a zodpovědnosti provozovatele a dozoru a jejich role a dále také ohledně schopnosti dozoru rohodnout nezávisle. Proto získaly otázky bezpečnostní kultury v souvislosti s licenčním procesem vysokou prioritu.

Povolovací řízení a spouštění pokračovala přesto, že důležité elementy takového procesu chybějí: nedestruktivní testy komponent primárního okruhu ještě nejsou dokončeny, neexistuje předprovozní analýza tepelného šoku, neodpovídající vyhodnocení zachování kontejnmentu v podmínkách havárií, analýza a řešení roztržení vysokoenergetických potrubí na podlaží +28,8 m je jenom marginální, enviromentální a seismická kvalifikace zařízení není pevně zavedená a funkční kvalifikace významných, pro bezpečnost důležitých komponent je stále otevřená, téměř žádné zahrnutí PSA studie, kterou provedla firma CEZ, jako podpůrný prostředek pro řadu rozhodnutí v bezpečnostní analýze) Severe Accident Management Guidelines se ještě musí vypracovat. Vede to k obavám, jak lze zachovat ochrany do hloubky vzhledem tomu, že důležité otázky kolem reaktorové nádoby, kontejnmentu a SAMG jsou ponechány nevyřešené během licenčního procesu. Rakouská strana je také znepokojená skutečností, že během diskuse o bezpečnostních otázkách a souvisejících licenčních aspektech dozor v ani jednom případě nesehrál roli odlišnou od provozovatele. Oficiální zástupce státního dozoru neprojevil zájem o otevřenou evaluaci důležitých bezpečnostních problémů a návrhů – ať už se týkaly provozovatele nebo jeho vlastní organizace.

Evropská praxe na základě nejnovějších postupů by nepovolila provoz pro 1. blok ani pro 2. blok, ani zavezení paliva v Jaderné elektrárny Temelín před úplným vyřešením výše zmíněných bezpečnostních otázek, především integrity reaktorové nádoby, předprovozních nedestruktivních testů, environmentální a seismické kvalifikace zařízení, fyzické separace vysokoenergetických potrubí na podlaží +28,8 m, funkční kvalifikace ventilů a chování kontejnmentu za podmínek havárie s tavením aktivní zóny.

Z těchto důvodů ještě nelze prokázat, že by jaderná bezpečnost JETE odpovídala stavu techniky ve státech EU. Klíčové analýzy lze provést během jednoho roku; výsledky těchto analýz jsou předpokladem k odhadnutí (možná velké) materiálové a časové náročnosti nápravných opatření k řešení nejzávažnějších bezpečnostních otázek, jako jsou křehnutí reaktorové nádoby, vedení vysokoenergetických potrubí na podlaží +28,8 m, integrita kontejnmentu, a seismické bezpečnosti. Provedení požadovaných analýz má nejvyšší význam.

Zusammenfassung

Im "Protokoll der Verhandlungen zwischen der tschechischen und der österreichischen Regierung vertreten durch Premierminister Zeman und Bundeskanzler Schüssel unter Teilnahme von Kommissär Verheugen" vom 12. Dezember 2000 (Melker Protokoll) kamen beide Seiten mit Kommissionär Verheugen überein, einen “Trialog” zu führen, mit dem Ziel zu einem besseren gegenseitigen Verständnis hinsichtlich des KKW Temelin zu gelangen. Im Hinblick darauf ersuchten die Regierungen Österreichs und der Tschechischen Republik die Kommission eine Expertenmission unter trilateraler Beteiligung durchzuführen. Ziel dieser Mission war es, den Dialog zwischen den Regierungen Österreichs und der Tschechischen Republik zur Frage der nuklearen Sicherheit zu erleichtern.

Der vorliegende Bericht wurde gegen Ende dieser trilateralen Diskussion verfaßt. Er spiegelt die Position der für die österreichische Seite tätigen internationalen Experten zu 29 zu Beginn des Prozesses definierten Sicherheitsfragen wider, auf der Grundlage während des Trialoges und davor zugänglich gemachter Informationen. Die Sicherheitsfragen wurden gemäß dem Abkommen ”auf der Basis des Stands der Technik, der in den Mitgliedstaaten der Europäischen Union gegeben ist” untersucht und bewertet.

Einige Sicherheitsfragen können – im Hinblick auf die zur Verfügung gestellten Informationen - als im Rahmen des Melker Prozesses abgeschlossen betrachtet werden. Andere wurden zur weiteren Diskussion auf die bilaterale Ebene oder die laufende Umweltverträglichkeitsprüfung (Melker Protokoll, Kapitel V) verschoben. Für eine Anzahl von Fragen liegen jedoch überzeugende Beweise für klare und schwerwiegende Abweichungen von Europäischen Praktiken und Standards vor, verschärft durch die Tatsache, daß diese Probleme nicht vor dem nuklearen Anfahren von Block 1 gelöst wurden. Obwohl einige dieser Probleme von der tschechischen Seite teilweise behandelt wurden bzw. in Aussicht gestellt wurde, diese zu einem späteren Zeitpunkt zu behandeln, besteht weiterhin ein klarer Bedarf für dringende Korrekturmaßnahmen.

Für die Österreichische Seite sind die wichtigsten Problembereiche in Zusammenhang mit den offenen Sicherheitsfragen:

1. <u>Keine Thermoschock-Analyse für Block 1 vor Anfahren des Reaktors</u>

Für Block 1 wurde keine vorbetriebliche Sprödbruchsicherheitsanalyse des Reaktordruckbehälters bei Themoschockbelastung unter Druck durchgeführt. Die stattdessen angebotenen vereinfachten betrieblichen Druck-Temperatur-Grenzkurven sind kein angemessener Ersatz für einen Sprödbruchsicherheitsnachweis, wie dieser von allen einschlägigen Codes gefordert wird.

Die Sprödbruchsicherheit des Druckgefäßes ist von höchster Wichtigkeit, da die Sicherheitssysteme weder in Temelin noch anderswo dafür ausgelegt sind, plötzliches Versagen eines Reaktordruckbehälters zu beherrschen. Ein Störfall dieser Art könnte zu Freisetzungen von radioaktivem Material in die Umwelt führen, die sich unmittelbar auf Österreich auswirken könnten. Die vorbetriebliche Sprödbruchsicherheitsanalyse dient dem Nachweis, daß während der gesamten Betriebsdauer trotz unvermeidlicher Materialversprödung ein hinreichend großer Sicherheitsabstand gegenüber katastrophalem Versagen des Reaktordruckbehälters unter Druck- und Thermoschockbelastung erhalten bleibt. Das Ergebnis kann erheblichen Einfluß auf den Lizenzierungsprozess einer Anlage nehmen. Aufgrund des hohen Nickelgehaltes des Druckbehälterstahls könnte das KKW Temelín unter den derzeit vorgesehenen Anlagenbedingungen (z. B. Neutronenbelastung) diesem Sicherheitskriterium nicht entsprechen. Einschneidende, schon mit Betriebsbeginn einzuführende Maßnahmen, wie die Neukonfiguration des Kernes, könnten erforderlich sein. Schon vor Betriebsbeginn auf thermisches Ausheilen – eine umstrittene Maßnahme zur Lebensdauerverlängerung – zu setzen, wäre mit den heutigen Europäischen Sicherheitsstandards unvereinbar. Die Tatsache, dass die Genehmigungsbehörde einem Zeitplan zugestimmt hat, welcher eine Sprödbruchsicherheitsanalyse erst innerhalb der nächsten fünf Jahre vorsieht und damit ein wesentliches Kriterium der Lizenzierung umgeht, wirft ernste Fragen zur Haltung der Beteiligten hinsichtlich Sicherheitskultur auf.

2. Konzept der zerstörungsfreien Werkstoffprüfung entspricht nicht dem Stand der Technik

Die zerstörungsfreien Werkstoffprüfverfahren für die Primärkreislaufkomponenten wurden noch nicht validiert (kalibriert), obwohl die zu diesem Zweck erforderlichen Testblöcke bereits zur Verfügung stünden. Aufgrund unzureichender Prüfung unerkannt gebliebene Materialfehler könnten die Materialfestigkeit und damit Unversehrtheit sicherheitsrelevanter Komponenten gefährden.

Spezielle Ultraschallprüfungsverfahren (in diesem Fall Tandem- oder französische Fokussierungstechnologie), die eine zuverlässige Feststellung von normal zur Oberfläche liegenden, rißähnlichen Defekten am Reaktordruckbehälter gewährleisten, wurden nicht angewandt.

In der Komponentendokumentation wurden Hinweise auf drei gemäß Normen unzulässige Defekte entdeckt, welche ohne die vorgeschriebenen Fehlerbewertungen belassen wurden.

Obwohl der Block 1 bereits in Betrieb genommen wurde, verfügt das Kraftwerk nicht über einen zusammenfassenden Bericht der Ergebnisse der durchgeführten Ultraschalltests des Primärkreislaufes. Ein zusammenfassender Bericht über sämtliche vorbetriebliche Prüfungsergebnisse – wie er von den Richtlinien der Genehmigungsbehörde gefordert wird – existiert auch nicht.

Die Rundschweißnähte der Frischdampfleitung auf der +28,8 m-Bühne wurden lediglich mittels Gammadurchstrahlung überprüft. Die auf die Frischdampfleitungen aufgeschweißten Fixierungsplatten der Ausschlagsicherungen wurden nach dem Schweißvorgang nicht überprüft (lediglich eine Schweißprobe wurde untersucht), die standardgemäße Inspektion nach Inbetriebnahme wurde nur an einer von 16 Fixierungen durchgeführt.

3. Qualifizierung von Komponenten für Störfallbedingungen noch unvollständig

Die Qualifizierung sicherheitstechnisch bedeutender Komponenten für außergewöhnliche Umgebungsbedingungen, wie sie bei Störfällen und Erdbebenbelastung auftreten können, wurde nicht vollständig durchgeführt, obwohl sich das Kraftwerk bereits in der Phase der nuklearen Inbetriebsetzung befindet. Dies entspricht nicht den international anerkannten Sicherheitsbestimmungen, welche den Nachweis der Qualifizierung sicherheitsrelevanter Komponenten vor der Beladung mit Brennelementen vorsehen. Abweichungen müssen als Ausnahmen behandelt werden und erfordern eine Genehmigung nach speziellen Regeln der Behörde.

Offizielle Qualifizierungsberichte, die noch vor Brennstoffbeladung zu erstellen, zu überprüfen, abzunehmen und vor Ort zur Verfügung zu stellen sind, liegen nicht vor. Ebensowenig konnte ermittelt werden, welche Rechtfertigung die Betreibergesellschaft für die Erteilung der Genehmigung zur Brennstoffbeladung ohne abgeschlossene Qualifizierung von Komponenten vorbrachte bzw. welche Auflagen die Behörde vorschrieb.

Nähere Angaben zu dem derzeit in Temelin durchgeführten Qualifizierungsprogramm wären wünschenswert, um (a) den geplanten Zeitpunkt der Fertigstellung der Qualifizierung sicherheitsrelevanter Komponenten festzustellen, (b) die Lizenzierungsaspekte der späten Qualifizierung von Komponenten, einschließlich Management und Dokumentation, nachvollziehen zu können, und (c) Daten hinsichtlich der Qualifizierungsresultate bestimmter Komponenten zu erhalten.

4. Erdbebensicherheitsbewertung entspricht nicht dem Stand der Technik

Die Erdbebensicherheitsbewertung für das KKW Temelin entspricht offenbar nicht dem Stand der Technik. Es gibt Hinweise, wonach das gegenwärtig angenommene Bemessungsbeben von 0,1 g maximaler horizontaler Bodenbeschleunigung, bei dem ein sicheres Abschalten des Reaktors noch möglich ist, das tatsächliche Risiko unterschätzt.

In einer Entfernung von 5 bzw. 13 Kilometern vom Kernkraftwerk verlaufen zwei wichtige tektonische Störungen (Jachymov/Hluboka und Blanice), einige weniger bedeutsame verlaufen sogar in noch geringerer Entfernung. Tschechische Experten berufen sich auf die Inaktivität dieser Störungen, zum Nachweis wurden jedoch noch keine dem Stand der Technik entsprechenden Datierungsmethoden angewendet.

5. Auslegungsmängel der +28,8 m-Bühne

Frischdampf- und Speisewasserleitungen verlaufen auf der +28,8 m-Bühne zwischen den Containment-Durchführungen und den Hauptabsperrventilen über eine Länge von über 20 Metern nebeneinander. Sie sind nicht physisch voneinander bzw. von anderen sicherheitsrelevanten Komponenten getrennt (beispielsweise durch Betonwände). Im Falle eines Bruches einer solchen Leitung sind Folgeschäden an benachbarten Leitungen sowie anderen sicherheitsrelevanten Komponenten infolge Rohrauschlages oder wegfliegender Bruchstücke nicht auszuschließen. Dies könnte ebenfalls einen schweren Radioaktivitätsunfall mit großer Freisetzung auslösen. Dieses Problem wurde nicht ausreichend behandelt.
Um die Tragweite der Auswirkungen im Falle des gleichzeitigen Bruches mehrerer Leitungen einschätzen zu können, bedarf es umfassender Analysen. Sie müssen detaillierte Informationen liefern betreffend Reaktivitätskontrolle bei solchen Unfällen, Abkühlung des Reaktordruckbehälters und der betroffenen Dampferzeuger und das damit verbundene Risiko eines Thermoschocks. Die Darstellung eines kompletten Störfallszenarios mit sämtlichen diesen Problembereich betreffenden Aspekten ist unbedingt erforderlich, welche u. a. die Bestimmung aller potentiellen internen und externen auslösenden Faktoren, Wasserschlag und dynamische Effekte in

Rohrleitungen sowie die Bestimmung zulässiger, begrenzender Bruchbedingungen umfassen sollte. Darüberhinaus muß dieses Störfallszenario in Aussicht genommene Rekonstruktionen samt Umgruppierung bzw. Verlegung sicherheitsrelevanter Komponenten und Rohrleitungen miteinbeziehen. Die Zweckdienlichkeit der am Durchtritt aus dem Containment und an der Trennwand zum Maschinenhaus installierten Ausschlagsicherungen muß ebenso wie der Nachweis vorbeugender Maßnahmen gegen Erosion und Korrosion einbezogen werden.

Hauptziel einer entsprechenden Neubewertung und Rekonstruktion an der +28,8 m-Bühne muß es sein, mehrfache Leitungsbrüche samt Folgeschäden an Komponenten und Ausrüstung, die durch die Sicherheitssysteme nicht kompensiert werden und daher zu schweren Unfällen mit großen Radioaktivitätsfreisetzungen führen können, technisch auszuschließen.

6. Funktionale Qualifizierung von sicherheitsrelevanten Ventilen offen

Für die Abblase- und Sicherheitsventile der Frischdampfleitungen ist die Qualifizierung der Funktionstüchtigkeit noch offen. Nicht qualifizierte Ventile könnten im Störfall bei Durchströmen von Wasserdampfgemisch in geöffnetem Zustand blockieren, wodurch letztlich ein schwerer Unfall mit Freisetzung von Radioaktivität ausgelöst werden könnte. Darüber hinaus fehlen Absperrventile in den Frischdampfleitungen der Anlage von Temelin, wodurch die nachteiligen Folgen blockierter Abblaseventile hintangehalten werden könnten.

7. Containment-Integrität nicht gewährleistet

Mehrere von der tschechischen Seite nicht oder nicht ausreichend behandelte Problembereiche wurden identifiziert, in denen Versagen des Containments im Falle schwerer Unfälle nicht auszuschließen ist:

Wasserstoff-Problem: Es wurden zwar passive autokatalytische Rekombinatoren (PAR) für bei schweren Unfällen entstehenden Wasserstoff installiert, allerdings sind Dimensionierung und Positionierung nicht für Bedingungen schwerer Unfälle ausgelegt. Zusätzliche, dem Stand der Technik entsprechende Analysen und Verbesserungen sind erforderlich.

Hochdruck-Kernschmelzauswurf: Hochdruckauswürfe von durch Kernschmelze verursachten Rückständen aus dem Reaktordruckbehälter können schwere Containmentleckagen nach sich ziehen. Der Einbau eines Druckunterdrückungssystems und die Analyse der mechanischen und thermischen Konsequenzen eines derartigen Unfallablaufes sind erforderlich.

Durchschmelzen der Reaktorgrube: Interaktion zwischen Kernschmelze und Beton können das Durchschmelzen der Reaktorgrube, die Freisetzung von Kernrückständen und das Eindringen luftgetragenen radioaktiven Materials in das Reaktorgebäude und nachfolgende Freisetzung in die Umwelt zur Folge haben. Analysen möglicher Modifikationen des Reaktorgrubentores sowie Kühlmöglichkeiten der Kernschmelze sind erforderlich.

Verzögertes Überdruckversagen des Containments: Länger andauernder Druckaufbau unter schweren Unfallbedingungen kann Überdruckversagen des

Containments nach sich ziehen. Einbau eines Druckablaßsystems mit Filter (Filtered Venting) wäre als Beitrag zur Lösung dieses Problems vorstellbar.

Containment-Bypass Unfälle: Zwei Ereignisabläufe, die zu Containment-Bypass führen (ausgelöst durch Brüche von Dampferzeugerheizrohren bzw. Leckagen von Dampferzeugersammlern) tragen zu ca. 2/3 zur Häufigkeit von Kernschäden, und damit möglicherweise schweren Unfällen, bei. Europäische Standards verlangen, daß Bypass-Unfälle nicht die dominante Ursache für Kernschaden darstellen.

8. Richtlinien zum Management schwerer Unfälle (SAMGs) nicht verfügbar

Die Implementierung von Richtlinien zum Management schwerer Unfälle (SAMGs) als Voraussetzung für adäquate "Defence-in-Depth"-Vorkehrungen gilt als westlicher Sicherheitsstandard. Für Temelin wurden solche Richtlinien noch nicht entwickelt und implementiert, obwohl CEZ bestätigt, daß SAMGs eine Notwendigkeit sind und obwohl diese für Temelin besondere Bedeutsamkeit haben, da diese Anlage den ersten WWER-1000-Reaktor mit einem hohen Anteil an Fremdtechnologie darstellt.

9. Sicherheitskultur

Auslegung, Bau und Betrieb der Anlage in angemessener Form auf der einen Seite, finanzielle Stabilität, Kompetenz und Verpflichtung des Betreibers und der Aufsichtsbehörde der Nuklearen Sicherheit gegenüber sowie Durchsetzungsfähigkeit und Unabhängigkeit der letzteren auf der anderen Seite sind Voraussetzungen für ein hohes Sicherheitsniveau beim Betrieb einer Nuklearanlage.

Die in den trilateralen Besprechungen und Workshops des Melker Prozesses behandelten Probleme waren ursprünglich vorwiegend auf der technischen Ebene angesiedelt. Im Laufe der Präsentationen und Diskussionen kam es jedoch immer wieder zu Situationen, welche Bedenken hinsichtlich der Verpflichtung von Betreiber und Aufsichtsbehörde zur Nuklearen Sicherheit und des Durchsetzungsvermögens der Aufsichtsbehörde aufkommen ließen. Die Frage der Sicherheitskultur im Lizenzierungsprozess gewann dadurch hohe Priorität.

Genehmigungsverfahren und Inbetriebsetzung erfolgten trotz Fehlen wichtiger Elemente dieses Prozesses: Unvollständige zerstörungsfreie Prüfung wichtiger Primärkreislaufkomponenten, nicht durchgeführte vorbetriebliche Sprödbruchsicherheitsanalyse, inadäquate Analyse des Containmentverhaltens bei schweren Störfällen, unzureichende Analyse und Lösung des Problems der Führung hochbeanspruchter Rohrleitungen auf der +28,8 m-Bühne, unvollständige Qualifizierung von Komponenten unter außergewöhnlichen Umgebungsbedingungen und unter Erdbebenbelastung sowie noch offene Qualifizierung der Funktionstüchtigkeit sicherheitsrelevanter Komponenten, weitgehende Nichtberücksichtigung der von CEZ erstellten PSA-Studie als Entscheidungshilfe bei Sicherheitsbeurteilungen sowie Fehlen der Richtlinien zum Management schwerer Unfälle (SAMGs). Bezüglich der Umsetzung des mehrstufigen Barrierenkonzeptes bestehen Bedenken, da wichtige Fragen betreffend Reaktordruckbehälter, Containment und SAMGs im Genehmigungsprozeß ungelöst blieben. Die österreichische Seite ist auch darüber beunruhigt, daß während der Diskussion der Sicherheitsfragen und der damit verbundenen Genehmigungsaspekte die Aufsichtsbehörde in keinem Fall ein sich vom Betreiber absetzendes Verhalten erkennen ließ; seitens des offiziellen Vertreters der Aufsichtsbehörde zeigte sich keinerlei erkennbare Bereitschaft, wichtige Sicherheitsbedenken oder Vorschläge - weder an die Betreibergesellschaft, noch an die eigene Organisation gerichtete – unvoreingenommen aufzugreifen und zu bewerten.

Nach europäischer Genehmigungspraxis dürften weder Temelin Block 1 noch Block 2 betrieben oder auch nur mit Brennstoff beladen werden, bevor oben genannte Unsicherheiten insbesondere betreffend die Integrität des Reaktordruckbehälters, die vorbetriebliche zerstörungsfreie Werkstoffprüfung, die Qualifizierung sicherheitsrelevanter Komponenten für außergewöhnliche Umgebungsbedingungen und seismische Belastungen, die Führung hochbeanspruchter Rohrleitungen der +28,8 m-Bühne, die funktionale Qualifizierung von Ventilen und Containmentverhalten bei Kernschmelzstörfällen beseitigt sind.

Es kann somit bislang nicht als nachgewiesen betrachtet werden, daß die nukleare Sicherheit des KKW Temelín dem Stand der Technik, wie er in den Mitgliedstaaten der Europäischen Union zur Anwendung kommt, entspricht. Die erforderlichen Analysen könnten zwar innerhalb eines Jahres durchgeführt werden; um allerdings zu den Schlüsselfragen Reaktordruckbehälterversprödung, Leitungsführung am +28,8 m-Niveau, Containmentintegrität und Erdbebensicherheit den (möglicherweise erheblichen) materiellen und zeitlichen Aufwand an Nachrüstungsmaßnahmen abschätzen zu können, ist das Vorliegen der Ergebnisse der angesprochenen Analysen unabdingbar. Der Durchführung der Analysen kommt somit vorrangige Bedeutung zu.

1 Introduction

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1.1 The Melk Process

With the "Protocol of the negotiations between the Czech and the Austrian Government led by Prime Minister Zeman and Federal Chancellor Schüssel with the participation of Commissioner Verheugen" of 12 December 2000 (the Melk Protocol) both sides agreed with Commissioner Verheugen in Chapter IV "Safety Issues" to conduct a "trialogue" to find a better mutual understanding on the issue of the Temelin Nuclear Power Plant. To this end, the Governments of Austria and the Czech Republic requested the Commission to undertake an expert mission with trilateral participation. The aim of this mission was to facilitate the dialogue between the Governments of Austria and the Czech Republic on the issue of nuclear safety.

Verbatim, Chapter IV. Safety Issues, states:

1. *Both sides agree with Commissioner Verheugen to conduct a "trialogue" to find a better mutual understanding on the issue of the Temelin Nuclear Power Plant.*
2. *To this end, the Government of Austria and the Czech Republic request the Commission to undertake an expert mission with trilateral participation. The aim of this mission is to facilitate the dialogue between the Governments of Austria and the Czech Republic on the issue of nuclear safety.*
3. *The expert mission of the Commission will be dispatched to Vienna to identify the main issues of concern with regard to nuclear safety at the Temelin NPP.*
4. *The same expert mission will hear the explanations given by representatives of the Czech Republic on these issues of concern, during a subsequent mission to Prague and the Temelin NPP.*
5. *The next step is a joint meeting to find solutions to identified problems on the basis of the state-of-the-art relevant in the Member States of the European Union.*
6. *The Austrian and Czech Governments agree to support the work of the ad-hoc Working Party of the Atomic Questions' Group by making available to the Council all already existing information, in particular any information resulting from activities under the bilateral agreements between the Czech Republic and the Federal Republic of Germany as well as the Republic of Austria, to be used at the discretion of this ad-hoc Working Party.*

During the first of the three meetings of the expert mission with trilateral participation on 2 February 2001 in Vienna the Austrian side presented 29 issues of concern regarding the safety of Temelin NPP on the basis of information and documentation that had been available before the Melk process. A summary of these concerns was published in the „*Austrian Report to the expert mission with trilateral participation according to chapter IV of the Protocol of the negotiations between the Czech and the Austrian Government led by Prime Minister Zeman and Federal Chancellor Schüssel with the participation of Commissioner Verheugen*“ (January 2001). As the Austrian concerns are primarily related to possible impacts of the operation of NPP Temelin on Austria, the issues are all directly or indirectly related to severe accidents.

The Czech side replied to the Austrian concerns during the second meeting within the Melk process on March 15 and 16, 2001, in Prague and Temelin, and summarised their reply in the “*Czech Republic Report to the expert mission with trilateral participation according to chapter IV of the Protocol of the negotiations between the Czech and the Austrian Government led by Prime Minister Zeman and Federal Chancellor Schüssel with the participation of Commissioner Verheugen*" (March 2001).

Two additional workshops were organised within the Melk process: The first at Rez, February 26 and 27, 2001, and the second in Prague, April 5, 2001. Technical consultations also took place in parallel to the second meeting of the expert mission in Prague and Temelin on March 16, 2001.

After the visit of Members of the Austrian Parliament to Temelin NPP on October 4, 2000, the exchange of information on expert level increased significantly and was further enhanced by the Melk process. The international team of experts commissioned by Austria (Austrian Expert Team) had the opportunity to examine some of the documentation regarding Temelin NPP on site. In particular access to the POSAR[1] and the PSA was given and then maintained throughout the Melk process.

There were, however some restrictions influencing the work of the Austrian experts: The workshops and technical consultations were limited: in the 4 days available, extensive presentations were given, reducing the time for study of the extensive documentation - occasionally to less than one hour (e.g. Rez meeting February, 26/27, 2001). Copies of documents or parts of them were not made available, any vital information had to be copied by hand. Some documents considered by the Czech side to be protected by proprietary rights were not made available. For most issues not covered by the open workshops, no direct contact between Austrian and Czech experts took place.

Due to the time schedule of the Melk Protocol, the expert mission with trilateral participation decided to defer some of the 29 issues to further bilateral discussion or to the ongoing Environmental Impact Assessment (Chapter V of the Melk Protocol). Of the 29 issues some can now be regarded as closed - in view of the information received - for the purpose of the Melk process (Table 1).

[1] Access to the POSAR was given with the exception of a few parts, e. g. on core design, chapter 4.2. fuel, 4.3. nuclear characteristics, 4.4. thermohydraulic characteristics

1.2 Structure of the Report

The present report presents the technical position of the Austrian experts on those concerns that have become sufficiently transparent to form an opinion and that have been recognised to be of special importance (see Table 1). The position is based on a careful review and evaluation of the information supplied by the Czech side as well as on relevant rules, guidelines and practices in the European Union and on the international level (e.g. IAEA).

In agreement with the Melk Protocol, which states the aim "… to find solutions to identified problems on the basis of the state-of-the-art relevant in the Member States of the European Union. …" (Melk Protocol, Chapter IV, p. 5), the subsequent chapters devoted to individual issues or clusters of issues describe

- the identified problem or deviation from the state-of-the-art and European practice,
- the possible solution or the further procedure suggested to reach a solution at a future time,
- the state-of-the-art relevant in the Member States of the European Union or documented in international guidelines and
- the significance of the deviation.

Where ever appropriate or necessary additional technical arguments are included.

1.3 State-of-the-Art relevant in Member States of the European Union

The EU *per se* does not have nuclear safety regulations — rather, each of the member states with operating nuclear power plants has its own regulatory structure, legal basis, regulations, guidance, citations to standards, etc. A common understanding of the “state-of-the-art relevant in member states of the European Union (EU)” has recently evolved. Based on this a clear hierarchy–approach was established for the purpose of this report. Additionally, there are four important considerations:

- The concept of the state-of-the-art relevant in member states of the EU is not limited to formal regulatory decisions. What is also highly relevant is what the practices, procedures, designs, and operating modes actually are at PWRs in member states of the EU (and secondarily, elsewhere outside the EU).
- In view of the limited number of nuclear power plants licensed in Europe in the last decade (see Table 2), it cannot just be initial licensing decisions which are relevant, but also re-licensing and license extension decisions. In member states of the EU, such decisions are often part of a Periodic Safety Review (PSR) process.
- Requirements issued after licensing are also relevant to the state-of-the-art.
- By definition, state-of-the-art relevant refers to advanced standards and practices, not the least common denominator of the relevant group of countries (e.g., member states of the EU). A specific example illustrates this: Not all the member states of the EU have required filtered venting systems (e.g., the regulatory authorities of Spain and the United Kingdom have not required filtered venting), however more than three-quarters of the PWR NPPs within the member states of the EU and

Switzerland have filtered venting systems. Under our concept of the state-of-the-art relevant in member states of the EU, filtered venting systems are "state-of-the-art".

The state-of-the-art hierarchy begins with practices and requirements derived from licensing and regulatory decisions in member states of the EU and continues downward to licensing practices outside Europe:

1. Practices and requirements derived from documents issued on European level
2. Practices and requirements derived from licensing and regulatory decisions in Member States of the EU
3. Actual practice at PWR NPPs in Member States of the EU
4. Practices and requirements derived from licensing and regulatory decisions in Switzerland
5. Actual practice at PWR NPPs in Switzerland
6. Practices within the Member States of the OECD Nuclear Energy Agency (NEA)
7. International consensus practices (including the International Atomic Energy Agency and international standards organisations)
8. Practices and requirements in individual non-EU European countries other than Switzerland
9. Practices and requirements in individual countries outside Europe

1.4 Urgency

European state-of-the-art practice would not permit plant operation before resolution of essential issues presented in this report. This implies that the resolution of all issues addressed here is urgent as Unit 1 of Temelin NPP is already in an advanced phase of start up testing up to 55 % of the nominal power output. Guidance was therefore sought on the time necessary to resolve the identified problems.

1.5 The Austrian Expert Team

The team of experts involved in the expert mission with trilateral participation and in the support of the mission comprised experts from Austria, Bulgaria, Germany, Italy, Russia and the USA:

Expert mission with trilateral participation			
Andreas **Molin**	AUT	Director, Division of Nuclear Co-ordination; Federal Ministry of Agriculture, Forestry, Environment and Water Management	Head of Delegation
Manfred **Heindler**	AUT	Consultant, Technical University of Graz, Member of the Nuclear Advisory Board	Nuclear Physics
Helmut **Karwat**	GER	Consultant, also to NEA/OECD and the European Commission	Confinement integrity
Norbert **Meyer**	GER	Consultant, Former Energie Nord Greifswald NPP, at present GARIF Consulting, Greifswald	Materials, weldments, reactor vessel embrittlement, NDT
Emmerich **Seidelberger**	AUT	Formerly Siemens KWU, Erlangen, Germany, and AE&E Waagner-Biro, Graz, Austria, at present Institute of Risk Research, Univ. of Vienna	Thermal hydraulics, confinement performance, accident analysis
Ilse **Tweer**	GER	Technical Consultant	Materials, NDT
Support Group:			
Iouli **Andreev**	RF	Consultant, Vienna, Austria, Former Technical Director of Spetsatom Emergency Centre in Chernobyl,	Emergency preparedness, external hazards
Harold **Denton**	USA	Consultant, Former Director, Office of Nuclear Reactor Regulation, NRC, at present Nuclear Safety Consultant, Knoxville, Tennessee	Nuclear safety regulation, licensing, safety analysis
Peter **Hofer**	AUT	Institute of Risk Research, University of Vienna	PSA, Emergency planning, radiology
Gueorgui **Kastchiev**	BUL	Technical Consultant, Formerly Head of the Nuclear Regulatory Commission of Bulgaria	WWER 1000
Franz **Kohlbeck**	AUT	Consultant, Institute of Geophysics, Technical University of Vienna	Geosciences, seismic hazard
Wolfgang **Kromp**	AUT	Head, Institute of Risk Research, Univ. of Vienna, Member of the Austrian Nuclear Advisory Board, Austrian representative to the PHARE&TACIS Expert Group	Materials, interdisciplinary risk assessment
Helga **Kromp-Kolb**	AUT	Head, Institute of Meteorology and Physics, University of Agricultural Sciences, Vienna	Meteorology, coordination
Roman **Lahodynsky**	AUT	Institute of Risk Research, University of Vienna	Geosciences, seismic hazard
Antonio **Madonna**	ITA	Technical Consultant to the Institute of Risk Research, University of Vienna	Nuclear safety regulation, equipment qualification, safety culture
Steven **Sholly**	USA	Former Senior Engineer, Beta Corp. Int. Inc., and Lambright Tech. Assoc. Albuquerque, NM; presently Institute of Risk research, Univ. of Vienna	PSA, external hazards, accident analysis, spent fuel storage safety
Geert **Weimann**	AUT	Consultant, Austrian Research Centres Seibersdorf, Member of the Nuclear Advisory Board	Materials, confinement performance, safety analysis, OSART
Hermann **Wüstenberg**	GER	Consultant, Federal Institute for Materials Research and Testing, Ultrasonic Methods Laboratory, Berlin	Materials, NDT-methods

Table 1:

Treatment of Issues during the Melk process (1: WS Rez, 2: Prague, 3: WS Post Prague, 4: IAEA-OSART, 5: EIA) and present status (Open / closed). The last column (#) refers to the Chapter of this report that addresses the issue.

	ISSUES	1	2	3	4	5	Open	Closed	#
1	Containment bypass and primary-to-secondary (PRISE) leakage accidents			X			Cluster C		7
2	Natural Gas Pipeline Accidents and Their Effect on Temelín		X						-
3	Tornadoes					X			-
4	Containment Design and Arrangement			X			Cluster C		7
5	Probabilistic Safety Assessment and Severe Accidents						Cluster C		7
6	Emergency Operating Procedures EOPs & Severe Accident Management Guidelines (SAMGs)				X				8
7	Seismic Design and Seismic Hazard Assessment	Additional documents							5
8	Main Steam Line and Feedwater Line Breaks		X				Cluster H		6
9	Reactor Pressure Vessel Embrittlement and Pressurised Thermal Shock (PTS)	X							2
10	Main Steam Line Safety and Relief Valves Qualification for two phase water flow	x					Cluster H		6
11	Status of IAEA Safety Issues resolution								-
12	Safety Classification of Components								-
13	Control Rod Insertion								-
14	Sump Screen Blocking and Suction Line Integrity		X						-
15	Reactor Coolant Pump Seal Integrity		X						-
16	Hydrogen Control						Cluster C		7
17	Limited ECCS/Containment Spray Sump Volume								-
18	Boron Dilution								-
19	Environmental and Seismic Qualification of Equipment		X						4
20	Ventilation System and Habitability Aspects of Control Rooms								-
21	Instrumentation and Control (I&C) Reliability								-
22	Non-Destructive Testing (NDT)	X					Cluster N		3
23	Leak Before Break (LBB)						Cluster N		3
24	Conception of Safety Features								-
25	Design Basis Accident Analysis								-
26	Beyond Design Basis Accident Analysis					X	Cluster C		7
27	Safety Culture				X		Cluster SC		10
28	NPP Organisational Structure and Management of Licensing Activities				X		Cluster SC		10
29	Technical Basis for Temelín Emergency Planning Zones (EPZs)			X					9

***Clusters**: **C**: "Containment Integrity in Severe Accidents"*
***H**: "Main Steam Line"*
***N**: "Non-Destructive Testing"*
***SC**: "Safety Culture"*

Table 2:

Most recently licensed PWR units in EU Member States and Switzerland:

The only EU PWR NPPs licensed since 1990 are the Units of Civaux and Chooz in France and the Sizewell B PWR in the UK. All other PWR licensings predate 1990 in the EU.

Member State	PWR	Licensing date
France	Civaux & Chooz	1996-1999
United Kingdom	Sizewell B	1995
Germany	Neckar 2	1989
Spain	Vandellos Unit 2 & Trillo	1988
Belgium	Tihange Unit 3	1985
Sweden	Ringhals Unit 4	1983
Finland	Loviisa Unit 2	1981
Switzerland	Goesgen	1979
Netherlands	Borssele	1973

2 Reactor Pressure Vessel Embrittlement and Pressurised Thermal Shock - Issue 09

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2.1 Introduction

The safety systems of PWRs are not designed to compensate catastrophic failure of the RPV (reactor pressure vessel). Therefore the different national regulations require restrictive measures to avoid possible brittle failure of the RPV. Special attention is to be paid to material selection, design and manufacturing of this sensitive component as well as on material degrading parameters and load transients during operation (under normal and abnormal operational conditions).

RPV steels even with high initial fracture toughness may degrade considerably due to embrittlement during operation. Embrittlement is caused by different degradation mechanisms, among them of particular importance, neutron irradiation.

During operation, certain abnormal conditions could result in so called pressurised thermal shock, i.e. rapid cooling of sections of the hot and still pressurised RPV by injection of relatively cold emergency coolant. Brittle failure under pressurised thermal shock conditions and increasing neutron embrittlement during operation is generally considered to be the major threat to RPV integrity. Thus before operation (fuel loading) of the RPV a thorough pre-service structural integrity assessment for pressurised thermal shock (PTS) conditions has to be performed in order to determine from the start the existence of a sufficient safety margin.

The result of this assessment can have considerable influence on the licensing process of a plant. Incisive measures of design changes could become necessary. However, in the case of WWER-1000 important measures such as heating of the emergency core coolant or the implementation of fast-acting isolation valves in the fresh steam lines are already exhausted.

Essentially, the remaining possibility is to significantly reduce the radiation impact through rearrangement of the core configuration (positioning of dummy or burnt-up fuel assemblies at the core periphery, low leakage core). This might help to reduce neutron embrittlement and ensure a sufficient safety margin against brittle fracture for the 40 years of life. Preconditions for a success of the strategy are firstly, a pre-service PTS demonstrating that non-permissible embrittlement will be reached rather late in service life and secondly, the application of a low leakage core from the start of operation. This measure would have to be implemented at the beginning of operation, because within the first 5 operational years about 50 % of the total (with respect to an end-of-life at 40 years) embrittlement will occur. An analysis only at the end of this period is far too late for efficient measures and thus of limited value.

To count on thermal annealing of the RPV – a disputed exceptional measure to reduce embrittlement at an advanced stage and thus prolong service life – already at the time before first operation would be incompatible with European safety standards.

2.2 Identified problems

- In deviation from international and European code regulations and practice (for instance Germany, France, UK/Sizewell, and even Slovakia/Mochovce) a pre-service structural integrity assessment for pressurised thermal shock (PTS) conditions was not performed for the RPV of Temelín Unit 1. It is intended to perform a PTS analysis for both units only within the next 5 years.

- The Czech side justifies the delayed realisation of the PTS analysis by calculated p-T operational limiting curves (Westinghouse concept). But such a substitution is not permitted under any code. Besides, the p-T curve calculations cannot be considered to be conservative with respect to the temperature field because the development of cold plumes is not considered. Furthermore, Russian experiments indicate that the embrittlement coefficients as specified in the Russian Code might not be conservative.

- The Temelín RPV steel is susceptible to neutron embrittlement, especially due to the high Ni content. International practice (see INSAG-12, German basic safety requirement) asks for materials with low neutron embrittlement sensitivity. The applied Westinghouse core re-design does not foresee the use of dummy assemblies for neutron fluence reduction, although the neutron fluence is high and the IAEA [1] in their WWER-1000 assessment gave a recommendation to consider this. Thus PTS analysis is of specific importance in Temelín NPP.

- The situation is aggravated for Temelín Unit-2, since the RPV material of Temelín Unit-2 has a higher initial brittleness temperature T_{k0} [1] than the one of Unit-1.

- Speculating with thermal annealing of the RPV – a disputed measure to prolong service life – already at the time before first operation would be incompatible with

[1] Brittleness temperature T_k is the reference temperature defined in the Russian Code describing the ductile-brittle transition of the material, T_{k0} is the initial brittleness temperature of the unirradiated material.

European safety standards. The fact that the licensing authority has agreed to a schedule, which provides PTS analysis only within the next 5 years and thus evades an important licensing prerogative raises serious questions regarding safety culture.

2.3 Solutions to identified problems

European regulations (KTA, RCC-M) and practice (Sizewell B) require a pre-service PTS (pressurised thermal shock) analysis.

Depending on the results of the analyses, measures such as the use of dummy or burnt-up fuel assemblies might need to be implemented from the beginning of operation. The need for corrective measures could be even higher for Unit 2 due to higher initial brittleness temperature

Timeline:
Neither for Temelin Unit 1 nor Unit 2 would European state-of-the-art practice permit operation or even fuel loading before finalising the pre-service PTS analysis. The analyses for both units could be accomplished within one year.

2.4 Deviation from State-of-the-art and Significance

The safety systems (emergency cooling circuit systems, containment) of PWRs are not designed to compensate brittle fracture caused catastrophic failure of the RPV. Therefore the different national regulations require restrictive measures to avoid possible brittle failure of the RPV under pressurised thermal shock conditions and increasing neutron embrittlement during operation. In order to ensure prevention of neutron induced brittle fracture through the RPV's entire service life, international state-of-the-art practice requires proof of a sufficient safety margin before start of operation. Proof is accomplished by the pre-service PTS analysis.

International codes and regulations ask for the use of materials with low neutron embrittlement susceptibility (see for instance German basic safety requirements or INSAG-12). The WWER-1000 RPV steel however is highly susceptible to neutron embrittlement (especially due to the high Ni content), and experimental investigations indicate non-conservatism of the Russian Code specifications used so far.

The German regulations limit the permissible EOL (end-of-life) neutron fluence to the RPV wall (KTA 3203/RSK Guidelines for PWRs: $1x10^{19}$ n/cm^2 E>1 MeV), other countries' codes require appropriate measures to reduce the neutron fluence at the RPV wall (this was also recommended for Temelín by the IAEA ("Safety issues and their ranking. WWER-1000 model 320 NPPs for Temelín NPP", 1999). Neither measure is adopted in Temelin, Unit 1 and 2.

According to general practice, the PTS analyses (pressurised thermal shock analysis) has to be performed for pre-service conditions, and has to be updated during operation according to valid boundary conditions and the results of the surveillance programme. The analysis has to consider all accident transients involving thermal shock load taking into account non-symmetric cold plumes and different postulated crack configurations in order to find the most stringent loading conditions. The results of these calculations tell

the safety margin during the lifetime of the RPV and thus give an important indication whether the unit's specific structural and operational conditions and the RPV steel are compatible. For instance, early measures might have to be realised with respect to the reduction of the neutron flux and/or the softening of the temperature shock at the RPV wall.
Thermal annealing of the RPV is a controversial measure to ameliorate embrittled base material and welds of the RPV wall. Up to now, the case of Loviisa, an old WWER-440 type plant in Finland, remains the unique case of annealing application among European member states. Efficiency of this process and adverse side effects of thermal annealing are controversially disputed. No validated experimental data are available for this exceptional measure, especially for the WWER-1000 steel [2] with a composition deviating from the WWER-400 steel that is more extensively investigated. Annealing was only introduced as a last resort when older plants, built at a time when information about embrittlement was limited, were endangered by early embrittlement. In any case no normative basis for relying on this technology during licensing exists in European member states, in Russia or in the Czech Republic. At present, the development of code regulations in Europe is evolving in the opposite direction[2]: Due to the large uncertainties involved, the above mentioned demonstration of structural integrity throughout the lifetime by PTS is required before first operation.

Recent experience of PWR PTS analysis calculations have shown that the use of verified thermal hydraulic codes indicate stronger thermal shock loads for the reactor pressure vessels than previously assumed (e.g. Mochovce NPP Unit-1).

No pre-service structural integrity assessment was performed for Unit-1 and Unit-2 of Temelín NPP. The Czech experts plan the realisation of a PTS analysis only within the next 5 years. This is certainly not in agreement with European practice. The Czech experts claim that the simplified Westinghouse concept for the calculation of operational p-T limiting curves would guarantee brittle fracture mitigation for the first 5 years of operation.

This procedure is not equivalent to European regulations and practices. Besides, the strongly simplified calculations are not conservative: the envelope accident transient is assumed to be a step-like temperature transient with rotational symmetry. Published calculations show that non-rotational symmetry (development of one or several cold plumes) yields significantly higher stresses and therefore stronger loads possibly causing brittle fracture.

Furthermore – should the PTS analyses require measures to reduce neutron fluence (to decrease neutron embrittlement), these measures would have to be implemented at the beginning of operation, because within the first 5 operational years about 50% of the total (with respect to an end-of-life at 40 years) embrittlement will occur. An analysis only at the end of this period is far to late for efficient measures and thus of limited value.

Therefore the pre-service PTS analysis (structural integrity assessment under pressurised thermal shock conditions) needs to be realised for both units, European standards and practice would not permit operation without this precondition.

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[2] According to international consensus (see for instance INSAG-12) neutron sensitive materials are not allowed for new NPPs.

Potential impacts on Austria: The structural integrity assessment and the reliable knowledge on irradiation induced degradation of the material toughness properties are required to determine the RPV lifetime and the safety margins during operational transients. Reactor pressure vessel failure could result in an early failure of the containment. A brittle RPV failure (circumferential rupture) could not only induce core meltdown but could also destroy supporting structures and the containment by RPV missiles. Such a severe accident could have a direct impact on Austria.

2.5 Technical Arguments

2.5.1 Introductory remarks on RPV structural integrity assessment

The assurance of the reactor pressure vessel (RPV) structural integrity throughout the lifetime is an important issue within the safety philosophy for pressurised water reactors (PWR).

The structural integrity assessment is important because sudden failure of the reactor pressure vessel due to brittle fracture cannot be compensated by the safety systems ([3], [4], [5]). Such an accident could result in release of radioactive materials into the environment.

National safety regulations include detailed prescriptions for the control of the brittle fracture safety. The safety documentation of nuclear power plants contains extensive evaluations based on specific material properties (fracture toughness, ductile-brittle transition temperature or specific reference brittleness temperatures, neutron embrittlement coefficients, etc.) and the specific loads on the component during pressurised thermal shock caused by cold water emergency injection in case of small LOCA (when the leak can be compensated) with simultaneous high pressure.

Therefore for new PWRs a complete pre-service structural integrity assessment has to be performed in order to demonstrate the brittle fracture safety of the RPV throughout the lifetime of the plant. Its importance is underlined by major structural shortcomings of the Temelín design such as described in the high energy line break issue (see Issue 8, chapter 5) which increase the probability of PTS events.

Embrittlement is caused by different degradation mechanisms, of which neutron irradiation is of particular importance. Due to neutron embrittlement of the RPV steel the brittle fracture safety diminishes with increasing operation time. Thus the process of neutron embrittlement has to be monitored during operation of the plant. In view of the consequences of failure the degree of embrittlement has to be quantified conservatively and the complete structural integrity assessment has to be performed with strong conservatism, maintaining considerable safety margins. Because of lack of measured data at the time of pre-service structural integrity assessment, conservative normative minimum specifications for the required material properties and very conservative assumptions on the postulated crack configurations must be introduced. After start-up of the power plant the brittle fracture safety assessment has to be updated on the basis of the surveillance programme results and has to consider indications of the non-destructive ISI (in-service) inspections; in this case the conservative assumptions on postulated cracks can be replaced by the real observed defects in accordance with the respective regulations.

2.5.2 European normative regulations

2.5.2.1 Russian Code

The Temelin NPP with two WWER-1000 units is fundamentally a Russian design plant (general designer of WWER-type reactors: OKB Gidropress, Podolsk) and was manufactured according to the Russian design and specifications by Skoda Plzen.

Based on this fact the Russian Code Regulations are of special importance for the normative regulations because they include the complete experience of design, construction and operation of WWER power plants. The new 1989 version of the materials strength standards [6] replaced the precursor norms dating from 1973, containing extensive changes and supplements especially with respect to the structural integrity assessment. This new code requires the realisation of a structural integrity assessment (point 1.2.1.(1) and 1.2.11.) for the licensing of a specific power plant. Paragraph 5.8. contains detailed instructions with tabulated data, including the T_k^a criterion that is used as standard methodology. Since then extensions and more precise instructions have been developed [7]:

- Improvement and qualification of calculation methods (determination of the thermal hydraulic parameters during an accident transient (e.g. RELAP Mod 5) and calculation of the mixing conditions with computer codes that were verified by large scale experiments, refinement of the calculations methods for temperature and stress fields, admissibility of other calculation methods for the stress fields)

- Regulations on postulated crack configurations with aspect ratios other than 2/3 and requirement of calculation of the stress intensity factor for the complete crack ligament (not only at the deepest point of the crack).

2.5.2.2 Germany: KTA regulations

The German KTA regulations require in KTA 3021.2 “Primary Circuit components of light water reactors; Part 2: Design, construction and calculations, topic 2: General principles”, Paragraph 5: “In correspondence to the safety relevant proposition of the components their structural safety, integrity and functionality has to be demonstrated.”

The component integrity is defined in Paragraph 5(b): “Integrity means that the pressure exposed wall has to reliably withstand all specified pressure and other mechanical loads in the frame of the specified loading frequencies and lifetime. The integrity is demonstrated by the strength assessment of the pressure exposed wall. “

Paragraph 4.6 “Irradiation” states:
“The neutron irradiation induced embrittlement has to be considered within the brittle fracture safety assessment.”

These general requirements are described more precisely in Paragraph 7.9 “brittle fracture analysis”. In 7.9.1. it is stated in (1): “The brittle fracture safety has to be demonstrated”.

And in (3): “Therefore it has to be demonstrated that the regions with possible irradiation embrittlement, esp. in case of large wall thickness and for high strength materials initiation of brittle fracture can be excluded in the heat affected zones and welds”.

And in (6): “For the demonstration of brittle fracture safety the concepts described in 7.9.2 or 7.9.3 have to be applied. It has to be taken into account that the neutron irradiation is increasing the ductile-brittle transition temperature during operation ...”

The methodology according to Paragraph 7.9.2 “Ductile-brittle transition temperature concept” by Pellini/Porse requires knowledge of the crack arrest temperatures that are not available for the WWER-1000 reactor in Temelín. Therefore the methodology described in 7.9.3, the fracture mechanical concept, has to be applied.

The cited passages from the KTA regulations allow the following conclusions:

- The brittle fracture safety has to be demonstrated within the design process (pre-service structural integrity assessment).

- The brittle fracture safety has to be demonstrated over the complete projected lifetime taking into account the neutron irradiation induced embrittlement.

2.5.2.3 French Code regulations

The French Code RCC-M “Design and construction rules for mechanical components for PWR nuclear islands” states in B3261: “It shall be verified that the loadings specified for the various conditions under consideration cannot cause fast fracture of the component, with material properties and defects which may exist in certain zones taken into account. This verification may be based on material properties and on analysis performed in accordance with guidelines below” and Z G 3230: Fast fracture analysis – level A criteria, Z G 3231: “The fast fracture resistance of ferritic steel components is evaluated for each zone for conditions selected from the set of conditions requiring compliance with level A criteria: the conditions selected shall constitute envelope conditions for the purposes of fast fracture analysis of the zone under consideration. ... any irradiation effects being taken into account in accordance with Z G 3430.”

2.5.2.4 IAEA Guidelines

The IAEA initiated in 1990 a programme to assist the countries of central and eastern Europe and the former Soviet Union in evaluating the safety of their nuclear power plants. The scope of the programme was extended in 1992 to include WWER-1000 plants in operation and under construction. The programme is thought as a forum to reach an international consensus on the technical basis for an improvement of the safety of these power plants (see introduction of [7]).

Within this programme as a result of long-term consultations “Guidelines on the pressurised thermal shock analysis of WWER nuclear power plants” were edited in 1997 [7]. These guidelines analyse the presently valid regulations (incl. [6]) and supplement the national norms by recommendations according to the international standard. Many international experts and organisations participated in this task:

- Brumovsky, M., Nuclear Research Institute, Czech Republic
- Cepcek, S., Nuclear Regulatory Authority, Slovak Republic
- Dragunov, Ju. G.; OKB „Gidropress“, Russia
- Elter, J., Paks Nuclear Power Plant, Hungary
- Faidy, C., Electricité de France, France
- Havel, R., IAEA
- Kovbacenko, C., Goskomatom, Ukraine
- Kovyrschin, V. G., Ministry for Environment and Reactor Safety, Ukraine
- Liska, P., Nuclear Power Plant Research Institute, Slovak Republic
- Matejovic, P., Nuclear Power Plant Research Institute, Slovak Republic
- Miannay, D., Institut de Protection et de la Sureté Nucléaire, France
- Miteva, R., Committee on the Peaceful Use of Atomic, Energy, Bulgaria
- Rantala, R., Finish Centre for Radiation and Nuclear Safety, Finland
- Rieg, C. Y., Electricite de France, France
- Sievers, J., Gesellschaft für Anlagen- und Reaktorsicherheit mbH, Germany
- Tendera, P., State Office for Nuclear Safety, Czech Republic
- Tuomisto, H., IVO International Ltd., Finland.

Considering this group of experts there is no doubt that the resulting Guidelines reflect the basics of the Russian Code [6] and the modern methodologies of the last 10 years from East and West. In addition, the participation of the Czech authors (from NRI Rez and SUJB) must be noted. Since this IAEA working group included national regulators and experts of the nuclear utilities it can be assumed that the guidelines reflect also a consensus between these groups of interest.

Thus, the IAEA Guidelines represent an adequate standard for the structural integrity assessment under pressurised thermal shock conditions of Russian design PWRs. These Guidelines were partially applied for the first time for the two units of the WWER-440/213 in NPP Mochovce. All the technical premises would be available to realise this concept also for the WWER-1000 of Temelin NPP.

2.5.3 Neutron embrittlement sensitivity of the steel

INSAG (International Nuclear Safety Group) provided in 1999 the basic safety principles for nuclear power plants [8]. For new power plants INSAG requires in Paragraph 216. for the core region: “Sensitive steels will not be used”. In the following this requirement will be discussed with respect to the expected neutron embrittlement of the RPV materials in Temelín NPP taking into account the development of Russian reactor steels for WWER reactors.

The construction of the reactor pressure vessels for the Russian WWER-type PWRs is based on a low-alloyed vanadium-stabilised chromium-nickel-molybdenum steel (12Ch2MFA; 15Ch2MFA; 15Ch2MNFA; Ch means chromium, M means molybdenum, N means nickel and F means vanadium), that was optimised with respect to its neutron embrittlement sensitivity in the course of industrial RPV manufacture.

The classification of reactor steels in non-sensitive, sensitive and highly sensitive is based on the shift of the ductile-brittle transition temperature due to the effect of neutron irradiation. Steels with a shift of this brittleness temperature below 50°C during the total plant lifetime of 40 years are considered to be relatively insensitive, steels showing a total shift between 50 and 100 °C are classified to be sensitive, in case of shifts above 100°C steels are classified to be very sensitive.

The neutron sensitivity is defined by the embrittlement coefficient A_F^T which determines the shift of the brittleness temperature at a certain irradiation temperature T due to irradiation with fast neutrons [6]:

$$T_K = T_{K0} + A_F^T * (\Phi * 10^{-22})^{1/3}$$

with:
- T_K: brittleness temperature
- T_{K0}: brittleness temperature in the unirradiated condition
- A_F^T: embrittlement coefficient at irradiation temperature T
- Φ: fast neutron fluence with $E \geq 0.5$ MeV in n/m^2

The table 1 shows the shift of the brittleness temperature for a reference fluence of 10^{24} n/m^2 for different neutron embrittlement coefficient. A_F^T:

A_F^T	ΔTin °C for $\Phi = 10^{24}$ n/m^2
9	42
10	46
11	51
12	56
13	60
14	65
15	70
16	74
17	79
18	84
19	88
20	93
21	97
22	102
23	107
24	111
25	116
26	121
27	125
28	130
29	135
30	139

According to the manufacturing documentation the first 70-MW reactors WWER-1 (Novo-Voronesh) and WWER-2 (Rheinsberg) were supposed to have embrittlement coefficients A_F^T between 9 and 12, thus the materials would have been classified as non-sensitive or slightly sensitive. Irradiation experiments in materials test reactors using samples from laboratory heats in the early 70's confirmed these assumptions. Therefore in the following years the reactors of the WWER-440 project had surveillance programmes. The evaluation of surveillance samples from the NPP Rheinsberg in the late 70's and beginning 80's revealed the first surprises: the measured shift of the brittleness temperature (ΔT_k) and thus the embrittlement coefficient were found to be significantly above the predicted values, which was very probably due to the so-called dose rate effect. Research programmes in the following years indicated that for identical fluence an irradiation with high neutron dose rates (neutron flux n/m^{2}s) yields lower shifts ΔT_k than irradiation with lower dose rate ([9], [10]). The irradiation in material test reactors is usually performed at high dose rates in order to gain information on irradiation effects of EOL (end-of-life) fluence within rather short time (irradiation times mostly below 1 year). The RPV materials are exposed to this fluence over the design life of 40 years - therefore the test reactor irradiation results have to be considered to be non-conservative. Due to this fact it is now required for surveillance programmes to use lead factors in the range of 2, in order to avoid a significant dose rate effect.

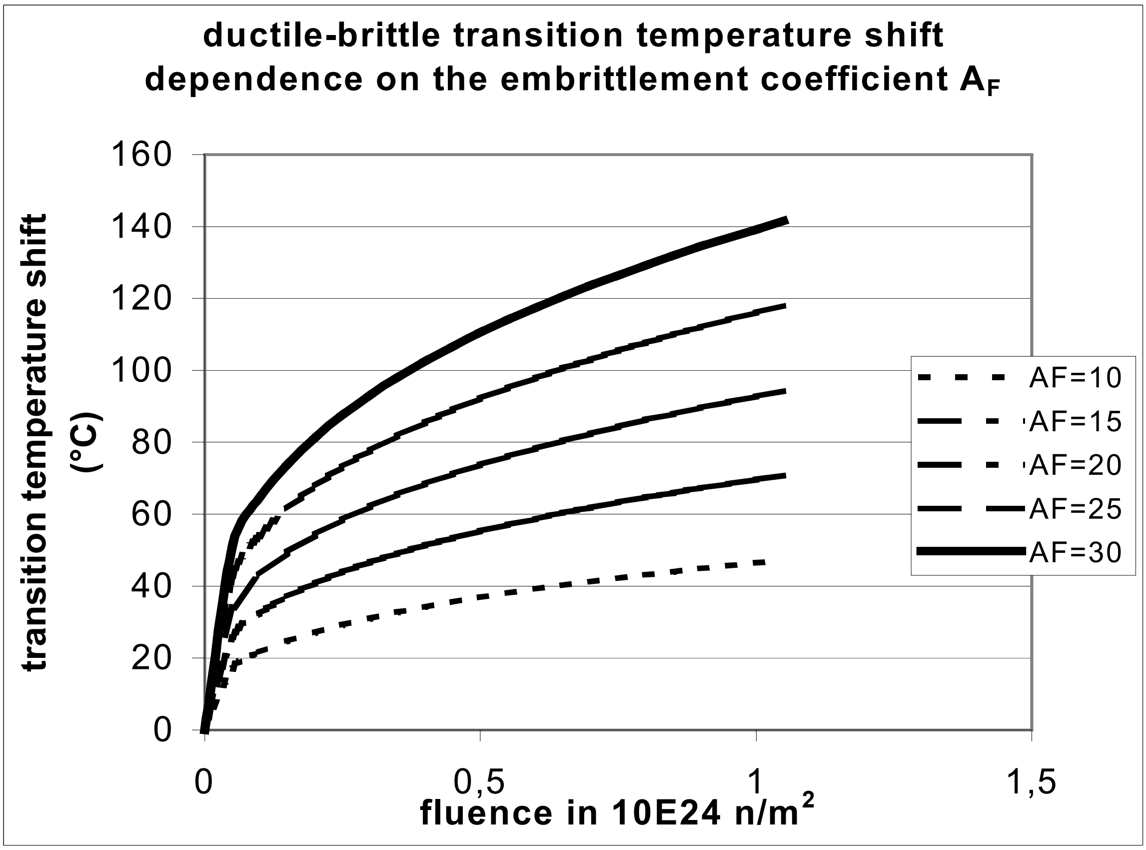

Figure 1 shows the increase of the brittleness temperature shift versus the fluence for different neutron embrittlement coefficients.

In consequence of the Rheinsberg results further irradiation experiments were performed using industrial reactor steels for the WWER-440 units. The results revealed another surprising effect: impurities of the steels or their weld materials (esp. copper and phosphorus) caused very high neutron embrittlement (coefficients partially over 20). Due to this effect the reactors of the WWER-440/230 project had severe problems handling neutron embrittlement. The fact that no surveillance programme was installed aggravated the situation. Extensive test series were performed to determine the correlation between copper and phosphorus content of the steels and the respective neutron embrittlement coefficient; finally this formula for the prediction of neutron embrittlement from the impurity content became part of the Russian Code. Due to the rather low-power emergency core cooling systems of these reactors maximum allowable brittlement temperatures up to 140°C were possible.

The last mitigation measure to avoid shut-down in case of brittleness temperatures surmounting the critical value is an annealing procedure (annealing of the RPV belt line region above 400 °C).

The requirement of reactor steel purity was fulfilled for the WWER-1000 type reactors: the respective steel 15 Ch2MNFAA; the restricted copper and phosphorus content of the base material and the meld metal was strongly controlled and observed. Later it was found that the increased nickel content of this steel caused a strong neutron embrittlement sensitivity even at irradiation temperatures of 290°C. Experimental test results on this steel showed embrittlement coefficients of 20 for the weld metal and 23 for the base material. These were implemented as predictive values in the Russian Code. The experiments were performed in materials test reactors at high dose rates, no results with low lead factors are yet available. The irradiation experiments performed in the test reactor Rez using original RPV materials of the Temelín NPP Unit-1 have also been realised with high lead factors (about 160). These experiments confirmed the Russian test results for high A_F^T values in the range of 20 or higher [11]. Therefore a warning was posted in POSAR, page 5.3-18 of Unit-1 [12]:

"Pursuing the RPV materials condition during operation with the surveillance programme it will be necessary to pay **maximum** attention to the **mechanical properties degradation of the weld metal** (weld no 3, RPV Unit-1)."

In the IAEA Guidelines for the pressurised thermal shock analysis [7] the European experts stated with respect to the WWER-1000 materials, that the normative embrittlement coefficient values of 20 and 23 are not conservative, if the nickel content of the steel is above 1.3%. The nickel content of the weld metal of Temelín Unit 1 is in the range 1.63 - 1.66%. The nickel content is also restricted according to other national regulations (KTA 3201.1: the maximum allowed Ni content is 0.85 %, US NRC Guidelines 1.99: design curves only up to 1.2 wt% Ni).

Recent experimental results also indicate that the embrittlement coefficients given in the Russian Code have to be considered to be non-conservative ([13], [14], [15], [16], [17]); published WWER-1000 surveillance data adjusted to an irradiation temperature of 290°C yield embrittlement coefficients A_F up to 41 (see attachment 1).

M. Brumovsky [18] reported on experiments made in former Czechoslovakia on neutron embrittlement of WWER steels, which showed that the irradiation induced shift of the fracture toughness - temperature curve is considerably higher (up to 60°C) than the ductile-brittle transition temperature shift deduced from notch tests:

„Experimental data has approved a suggestion that transition temperature shifts due to irradiation embrittlement, defined from notch toughness or from static fracture toughness temperature dependencies, are not equivalent. Analysis of recorded data is shown as well as its influence on RPV residual life assessment. Generally it was found that shifts in temperature dependencies of static fracture toughness are larger than for similar ones of dynamic fracture toughness as well as for Charpy impact tests."

Brumovsky concluded that this has important implications for future calculations of RPV service life. As no measurements of fracture toughness for irradiated surveillance test specimens exist, a considerable safety margin should be applied to assure a conservative evaluation.

Summarising the findings on the neutron embrittlement sensitivity of the WWER-1000 RPV steel in Temelín Unit 1, it has to be stated:

- The used RPV steel is very sensitive with respect to neutron embrittlement. The predictive embrittlement coefficients A_F^T in the Russian Code are 20 (weld metal) and 23 (base material).
- The data base for these code values is rather limited and based mainly on test reactor irradiation with high dose rates. Therefore it has to be expected that these embrittlement coefficients are not conservative.
- According to the IAEA Guidelines the code values have to be considered to be non-conservative due to the elevated nickel content above 1.3% in the RPV steel.
- The predicted neutron embrittlement based on the Russian Code values is therefore not conservative. It is recommended to use A_F^T values of 25 for the weld metal and 28 for the base material for a first prognosis of neutron effects in the pre-service PTS analysis.
- The surveillance programme for Temelín Units 1 and 2 are of extraordinary importance not only for the Temelín NPP but also for all WWER-1000 units, because this programme will yield the first reliable data base for a conservative evaluation of the neutron embrittlement sensitivity of these steels. It is therefore urgently recommended to accompany these experiments and evaluations by EU experts.

2.5.4 Constructive changes of the surveillance programme

The surveillance programmes realised in the hitherto existing WWER-1000 plants did not deliver reliable information on neutron embrittlement because of significant uncertainties within the determination of irradiation temperature and neutron fluence at the sample location. Due to the geometric arrangement of the surveillance capsules in the RPV no comparable neutron fluence could be achieved for a sample set in the capsules (neutron flux fluctuation within the capsule by a factor 5, e.g. in the range 2-10E10 n/cm^{2}s) [2]. The experimental determination of the embrittlement state of the RPV materials (base material, weld metal, heat affected zone) needs a set of up to 20 samples with identical neutron exposure for each material. Thus, no reliable information can be produced if the neutron exposure is not comparable within one set of samples. Furthermore, the surveillance capsule location close to the out-let nozzle caused an irradiation temperature in the range of 302-309°C, which is significantly higher than the temperature at the critical RPV weld in the active zone (290°C). For the same neutron fluence higher irradiation temperatures cause lower embrittlement compared to lower irradiation temperatures. Therefore the embrittlement values measured so far using WWER-1000 surveillance programme data are definitely to low to allow a realistic prediction on the embrittlement status of the critical RPV region. In addition, due to the lack of reliable neutron fluence data the hitherto available surveillance results are useless for a conservative prediction of the RPV structural integrity throughout the lifetime.

In order to get reliable results from the surveillance programme the construction principle for the surveillance sample containers has been changed for the Temelín NPP: 10 containers with new design are located close to the internal RPV surface assuring a lead factor of about 2 and irradiation temperatures close to the vessel wall temperature. The installation of new fluence monitors allows a reliable determination of the absolute neutron fluence and the fluence distribution in the containers. The installation of irradiation temperature monitors (low temperature melting alloy wires) according ASTM 1214-87 and KTA 3203 is planned after 2-year operation (information during the Trialog meeting in Prague, March 2001). In the mean time the irradiation temperature is apparently only estimated.

2.5.5 Critical assessment of the p-T operational limiting curve concept

A bilateral consultation on structural integrity assessment took place in Rez on September 14, 2000. The main issue of this meeting was the methodology for the PTS (pressurised thermal shock) analysis used for POSAR. The Czech expert declared that for Temelin NPP Unit-1 no PTS analysis was performed, neither according to the Russian Code from 1989 [6] nor according to the IAEA Guidelines from 1997 [7]. Only p-T operational limiting curves were calculated using the simplified Westinghouse concept. Based on these p-T curves the Czech experts are convinced that the operation of Unit-1 is safe for the near future and there is enough time to perform a PTS analysis within the next 5 years.

The simplified Westinghouse concept is not intended to replace the PTS analysis, but conclusions with respect to the necessity and the timing for a PTS analysis are drawn from these simplified calculations. Therefore a critical analysis of the concept is performed:

The aim of operational limiting curves is the determination of allowed and non-allowed regions in the pressure-temperature diagram, describing thermal shock-like loads with overcritical fracture mechanical stress fields. The topic is not the assessment of the RPV's structural integrity but the evaluation of allowable or non-allowable pressure-temperature conditions for all operational situations as additional information for the operator.

The analysis of accident transients is based on the following assumption: the pressure vessel is exposed to a shock-like step-function temperature transient with rotational symmetry with respect to the vessel belt line. The developing temperature gradients cause thermal stresses. The temperature and stress fields are calculated using modern calculation codes, taking into account the cladding effect for the temperature field. A heat transfer coefficient of 28,000 W/m^{2}K was assumed. The temperature drop was varied for several step heights down to the temperature of the emergency cooling water.

The postulated semi-elliptical cracks at the inner surface of the RPV (at the interface between cladding and ferritic vessel steel) were varied with respect to the crack depth from 5 mm to 48.125 mm (=1/4 wall thickness) using an aspect ratio of 2/3. For these postulated cracks the stress intensity factors were calculated using the Russian formula for the crack tip and the two points were the crack ligament reaches the inner surface. The latter ones turned out to be the most critical ones.

First, the stress intensity factors $K_I(T)$ are calculated for the temperature gradients in the pressure vessel wall, subsequently those internal pressure values of the primary circuit are determined for which $K_I = K_{Ic}$ by solving the equation $K_{Ic} = K_I(T)+p*F(K_I(p))$ with respect to p. F is the residual function from the calculation of the stress intensity factor due to the internal pressure load. The fracture toughness was described according to the Russian Code formula in [6]:

K_{Ic}=min{26+36 exp[0.02(T-T_k)];200}.

The resulting pressure-time diagrams combine all those values of internal pressure in one curve, for which the critical condition is valid over the accident transient. These curves are calculated for different temperature steps of the thermal shock load. The smallest pressure values for each temperature step that reach the stability criterion and the respective temperature yield the operational limit curve with the shock temperature on the x-axis and the internal pressure on the y-axis.

The calculations were performed at UJV Rez for T_k= 77°C and at Skoda for T_k values between –6°C and +38°C. The EOL (end of life) fluence is assumed to be 5.7E23 n/m^2 (E>0.5 MeV), the embrittlement coefficients (A_F) 20 for the weld metal and 23 for the base material. According to the Czech experts the resulting p-T limiting curves show that at least during the first years of operation no brittle fracture hazard should occur.

The Westinghouse concept of p-T operational limit curves is based on a shock-like temperature load (temperature transient of the primary circuit coolant as step function) bearing a high amount of conservatism according to the Czech side. Confirming calculations to support this statement would be needed, because:

1. The development of temperature gradients within the pressure vessel wall is mainly determined by the thermal conductivity of the steel. Those cases are critical where the radial temperature gradient causes bending stresses with tensile stress components deep into the vessel wall. Thus the sharpness of the temperature drop (thermal shock or cooling within minutes due to ECCS injection) is not necessarily decisive. In addition the thermal shock is damped by the soft austenitic cladding at the inner surface, therefore the step-like temperature drop does not include a very large conservatism. The level of conservatism can only be evaluated by comparing calculations with different postulated crack configurations.

2. The Westinghouse concept assumes a cylindrically symmetric temperature field in the pressure vessel. This does not reflect reality and therefore does not yield conservative loads. For several accident transients (ECCS injection, large leak in the secondary circuit) the development of one or more cold plumes has to be expected. This effect causes strongly asymmetric temperature fields in the vessel wall around the belt line. Additional stress components evolve that are not considered within the Westinghouse concept but can cause significantly higher loads depending on the crack location.

3. Especially in case of ECCS injection the cold plumes show a vertical temperature gradient along the vessel wall (warming of the water within the plume by mixing) that on the one hand reduces the shock load in the lower part but on the other hand causes an additional bending stress component in vertical direction. This stress component is especially dangerous for defects in the welds due to tensile stresses at the inner surface across the weld. This component is also not considered within the Westinghouse concept. It would have to be considered within a PTS analysis according to the IAEA Guidelines.

4. The calculations for the Westinghouse concept used an aspect ratio of 2/3 for the postulated crack configurations. The IAEA Guidelines require calculations with postulated crack aspect ratios up to 1/10 resulting in higher stress intensities. Therefore the performed calculations cannot be considered to be conservative.

Due to these arguments it is questionable as to whether the basis for the general statement that the Westinghouse concept is sufficiently conservative. It is possible that the conservatism of the step-like temperature function cannot compensate the non-consideration of the temperature gradients along the circumference and the height of the RPV. Calculations at GRS for WWER-1000 reactors [19] and Western PWRs [20] have shown that the loads in case of cold plumes are significantly higher than the thermal shock with rotational symmetry.

2.5.6 Urgency of a PTS analysis for Temelín Units 1 & 2

The structural integrity assessment for thermal shock conditions based on [7] includes several parts:

1. Determination of the parameter transients in the primary circuit for selected accidents for which a strong thermal shock is expected.
2. Calculations of the local mixing conditions in the downcomer in case of cold plume development (ECCS injection or large leak in the secondary circuit)
3. Calculation of the temperature fields in the pressure vessel wall over the complete accident transient
4. Calculation of the stress fields from the temperature gradients and the internal pressure over the complete accident transient
5. Calculation of the stress intensity coefficients for different postulated crack sizes and configurations over the total crack ligament (at least the deepest point and the points of intersection of the crack front with the boundary between cladding and base or weld metal) during the complete accident transient
6. Calculation of the crack loading paths (stress intensity factors versus temperature) for different crack configurations and all selected accident transients
7. Determination of the maximum allowable brittleness temperature T_k^a from the intersection of the load paths and the fracture toughness curve
8. Comparison of the T_k^a values with the neutron induced shift of the brittleness temperature T_k.

This procedure is international standard with slight differences in the national regulations with respect to the calculation methodologies. The IAEA Guidelines impose high requirements for the quality of the calculation codes, including the validation of calculation procedures by large-scale experiments. In the past modern validated calculation codes (esp. for (2)) for the PTS analysis of Russian WWER-reactors were not available. This situation has changed since 1990, but still only few complete PTS analyses for WWER plants were performed up to now. Such a PTS analysis using modern validated calculation codes was performed for the WWER-440 units of NPP Mochovce [21] following the methodology recommended in [7]. Unfortunately this analysis was not complete (non-conservative assumptions on the postulated crack configurations), but the results are nevertheless of importance for the WWER-440/213 reactors. Up to now T_k^a values in the range of 140°C have been published for WWER-440/230 reactors. The results for NPP Mochovce showed T_k^a values down to 73°C, although the analysis considered only small cracks with an aspect ratio of 2/3 and not the ¼ crack with extended aspect ratio as required by the IAEA Guidelines. Very probably even lower T_{ka}^{a} values (around 60°C) have to be expected.

First calculations for the WWER-1000 performed at GRS in 1995 [19] for small crack sizes (15 mm deep, 50 mm long) demonstrated that when assuming asymmetric cooling transients with cold plumes (200 cm^2 LOCA) the T_k^a value is smaller than 100°C, as compared to 135°C for rotational cooling symmetry.

The following initiating events are to be considered for WWER NPPs according to the recommendation of the IAEA ([7], attachment 4):

1. Spectrum of postulated piping break within the reactor coolant pressure boundary
2. Rupture of the line connecting the pressuriser and a pressuriser safety valve
3. Inadvertent opening of one pressuriser safety valve
4. Leaks from the primary to the secondary side of the steam generator
   SG tube rupture
   Primary collector leaks up to cover lift-up
5. Inadvertent opening of one check or isolation valve separating reactor coolant boundary and low pressure part of the system.
6. Inadvertent actuation of ECCS during power operation
7. Chemical and volume control system malfunction that increases reactor coolant inventory
8. Inadvertent opening of one steam generator safety or relief valve or turbine bypass valve
9. Spectrum of steam system piping break inside and outside of containment
10. Feedwater piping break

The maximum allowable brittle fracture transition temperature resulting from PTS analyses for Western-type RPVs (4-loop PWR) assuming a semi-elliptical surface crack (crack depth 16 mm, half crack length 48 mm) and asymmetric accident transients with cold plumes (200 cm^2 LOCA) was as low as 59°C [20]. The fracture mechanical evaluations did not consider safety factors. Therefore it is to be expected that T_k^a values in the range of 50-70°C might occur for WWER-1000 reactors.

2.5.7 Additional problems regarding structural integrity of the RPV

During review of selected documents (POSAR, reactor passport) two non-allowable indications (according to the accepted standards PK1514-72) were discovered in the RPV that were left unrepaired without the respective evaluation by technical calculations. A re-assessment of these indications should be performed according to the regulations, extensive observation including special procedures during ISI is necessary, in case growth during operation repair is unavoidable (see Issue 22, chapter 2).

2.5.8 Implications regarding Safety Culture

The facts that PTS analyses have been postponed, aggravated by the well known high susceptibility to neutron embrittlement of Temelín RPV, the incomplete NDT and detected non-allowable indications do not indicate a high level of safety culture on the part of the Operator.

According to international practice the commissioning procedure would not have proceeded to the extent it has in Temelín NPP without PTS analysis. There is reason for concern about the actions required by SUJB as the Regulator and the Licensing Authority as well.

2.5.9 Conclusions

- Due to the high irradiation sensibility of the steel there is - under the given conditions - a very high probability that brittle fracture safety will not be demonstrable throughout lifetime neither for RPV Unit-1 nor RPV Unit-2, if possible thermal shock loads caused by asymmetric temperature distributions (cold plumes) are considered.
- The brittle fracture hazard for selected accident situations can occur rather early, in extreme cases after 5 years of operation.
- The caveats regarding PTS analysis must be seen also in the context of the insufficiency of pre-service testing of the RPV and the unsatisfactory consideration of the results (see Issue 22: Non-destructive Testing).
- Thermal annealing of the RPV as a potential measure for life-time extension would not be acceptable. Besides the lack of data in the case of WWER-1000 steel annealing, this measure is controversial and speculation with annealing at the time before first operation would be incompatible with European safety standards.
- Partial mitigation of the neutron embrittlement can only be reached if measures are being taken to reduce the radial fluence component beginning from start-up.
- The first seven topics of the structural integrity assessment for conditions of pressurised thermal shock are mostly independent of materials properties. The calculations could and should therefore be commenced immediately, even more so, as the exercise is very time consuming. Because of the principal importance of this analysis for all WWER-1000 reactors external European experts should participate.
- As long as no reliable information on the neutron embrittlement is available conservative embrittlement coefficients should be used for (8).
- The investigation of the surveillance programme samples is of extreme importance, not only for Temelín NPP. This is also strongly recommended by WENRA 2000[3]. The evaluation of the samples should be promoted with international assistance.
- The handling of the issue by the Operator and by the regulatory authority raised concern regarding their understanding of safety culture.

[3] WENRA 2000 „Nuclear Safety in EU candidate countries“: (43): The high quality of the reactor pressure vessel, manufactured by Skoda, Plzen, is well documented. The Nickel impurity content, however, is somewhat higher than today’s more stringent specifications. To determine the effect of neutron irradiation on the material, a special irradiation programme covering end-of-life fluence condition has been performed. However, due to some uncertainties, a final assessment with regard to expected changes of material properties currently cannot be made. Therefore close attention has to be given to the monitoring of the embrittlement of the RPV during operation.

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2.7 Attachment 1: Published data on WWER-1000 neutron induced embrittlement

In KAMENOVA 2000 the given A_F-values for weld metal surveillance samples from different WWER 1000 plants are between 6 and 29, the lowest A_F values do not coincide with the lowest Ni content, neither with the lowest P or Cu content

Table 1: surveillance samples of WWER-1000 RPV weld metal

Reference	**NPP/Unit**		**Ni(%)**	**P(%)**	**Cu(%)**	**A_F**	**adj. A_F(290°C)**
Kamenova 99	Balakovo 1	weld	1.88	0.009	0.028	29.0	35
Kamenova 99	Kalinin 1	weld	1.76	0.01	0.040	35.0	41
Kamenova 99	Novovoronesh 5	weld	1.21	0.014	0.040	8.0	14
Kamenova 99	S Ukraine 1	weld	1.72	0.008	0.050	16.0	22
Kamenova 99	S Ukraine 2	weld	1.72	0.005	0.060	6.0	12
Kamenova 99	Bulgaria	weld	1.70	0.009	0.030	12.0	18
Boehmert 00	Archive/255°C	weld	1.71	0.04	0.012	47.5	33.5

The irradiation temperatures for surveillance-samples in WWER-1000 pressure vessels were according to Kamenova “not experimentally measured but expected to be in the range 305±5°C”, the RPV wall temperature at the critical circumferential weld is specified with 290°C. Thus the measured embrittlement for a certain fluence found in surveillance samples is due to the higher temperature certainly lower than the embrittlement of the belt-line weld material.

Using the formula given in VIEHRIG 1999
$A_F(T_{irr}) = A_F(T_v) + K \times (T_v - T_{irr})$ (K=0.2 for base metal and 0.4 for weld metal)
the respective adjusted A_F (290°C) can be calculated from data of another irradiation temperature T_{irr}; these values are given in the last column of table 1.

According to DAVIES 1999 the surveillance chains in WWER-1000 located “such that their irradiation temperature reflected the coolant outlet temperature (322°C) rather than the PV wall and this could introduce a lack of conservatism”. An adjustment using an irradiation temperature of 320°C would increase the A_F values by +6.

WWER-weld metal irradiated at 255°C in Rheinsberg showed A_F values of 47.5, the adjustment to the vessel temperature of 290°C gives 33.5 (see table 1, last row).
Temperature uncertainties of ±5°C change the A_F value by ±2
(higher temperature $\Rightarrow$ lower A_F).

The specification for WWER-1000 steels predicts A_F values (for 290°C) of 23 for base metal and 20 for weld metal.

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3 Cluster "Non-Destructive Testing"

Issue 22: Non-Destructive Testing
Issue 23: LBB

Table of Contents

3.1 Introduction

The aim of the NDT (Non-Destructive testing) assessment is the quality assurance of safety relevant components of the primary and secondary coolant circuit with respect to cracks and material defects that could jeopardise the strength and thus the component integrity.

Pre-service inspections are important for the evaluation of the quality level after manufacture and assembly of the NPP components and allow repair before the components are contaminated. The pre-service test results are baseline data for comparison of future developments. During the operational life of the plant in-service inspections (ISI) are run to monitor status and degradation of materials, and to ensure that ageing effects do not degrade safety margins.

3.2 Identified Problems

The volume and the quality of the non-destructive testing in the primary circuit in the commissioning phase do not meet European practices.

3.2.1 Non-Destructive Testing (NDT)

- The NDT programme has not yet been qualified. The NDT qualification programme for primary loop components should be finalised using the test blocks already available at NRI Rez (incl. ferritic pipes/austenitic cladding).
- No ultrasonic test methods (i.e. Tandem technique or French focussing technique) were used that guarantee reliable detection of severe crack-like defects, especially cracks perpendicular to the surface in the reactor pressure vessel (i.e. originating from defects

not detected during manufacture). The TOFD (Time Of Life Diffraction) technique is not internationally validated or PISC-recommended. The probability that defects relevant to safety are not detected (e.g. those oriented perpendicular to the surface and originating from defects not detected during manufacture) can therefore not be neglected.

3.2.2 Pre-service NDT (Non-destructive testing) documentation

- In spite of start-up of the plant and the requirements of Reg. 436 of the CAEC, Oct. 10, 1990 ([1]), § 5 (2) and § 8 (1)/(6) the final documents of the RPV ultrasonic testing after the hot water test and the mechanised ultrasonic testing of the primary circuit are not available at the plant. The original information from the mechanised testing is available only at the manufacturer of the components who is also responsible for the inspection.

- There is no summarising document on all non-destructive pre-service test results concerning the main components of the primary circuit and selected components of the secondary circuit (steam lines, steam generator collector). Such a document must include a listing of used methods, description of the methodology, discussion of the used sensitivities and comparison of the licensing and standards requirements with the actually performed testing, and a complete traceable listing of all registered indications, in particular those which have to be considered deviations and those requiring repair. Such a summarising document is an important reference for the Operator and the Regulator.

- During review of parts of the POSAR and the reactor passport - due to the restricted time schedule the complete documentation could not be reviewed - three non-allowable indications (according to the accepted standards) were discovered that were left unrepaired without the appropriate defect evaluations.

- The circumferential welds of the main steam lines on the +28.8 m level were tested using only X-ray radiography. Wedges[1] welded onto the main steam lines were not inspected after welding (only an acceptance test weld was inspected), ISI was performed on only one wedge. A mechanised UT procedure is not yet included in the ISI programme for the circumferential welds of the main steam lines. Complete ultrasonic examination including special test procedures enabling possible underweld crack detection at the wedges welded on to the main steam lines have not been performed in spite of the severe accident potential of the +28.8 m level (see Cluster “28.8 m Level").

- The Austrian concerns with respect to the status of leak-before-break (LBB) were not eliminated, possibly because the Austrian experts had no opportunity to evaluate additional documents on the LBB application besides the POSAR and a very early Czech LBB write-up document. In addition, there is concern about the credit taken from the LBB concept implementation as a means to justify accident load combinations deviating from presumably valid code requirements in the case of SSE (safe shut-down earthquake)/DBE (design base earthquake) events.

[1] These attachments, called "fixations" in the Czech documents, are wedges positioned circumferentially around the line and are intendend to provide firm connection beween line and whip restraint

3.3 Solution to Identified Problems

1. Non-destructive testing (NDT):

(a) Finalisation of NDT qualification programme (incl. test blocks for ferritic pipes/austenitic cladding)

(b) Inclusion of validated UT methods for reliable detection of crack-like defects perpendicular to surface in safety relevant components (RPV, RPV head and components with wall thickness over 100 mm). Validated and PISC-recommended are Tandem or French Focussing Techniques (TOFD is not internationally validated).

(c) Application of mechanised UT inspection to the main steam lines, and of special procedures for pipe whip restraint wedges which are welded to the main steam lines (underweld cracks).

2. Pre-service NDT documentation:

(a) Summarising report available at the plant, including required/performed testing volume, NDT methodology and sensitivity, listing of all indications above registration limit

(b) Re-assessment of the non-allowable indications according to the regulations (specific NDT programme, strength analysis)

Timeline:
Neither for Temelin Unit 1 nor Unit 2 would European state-of-the-art practice permit operation or even fuel loading before resolution of the NDT issue.
All necessary measures for Unit 1and Unit 2 can be accomplished within one year.

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

Pre-service non-destructive testing (NDT) are important for the evaluation of the quality level after manufacture and assembly of the NPP components. Pre-service NDT as a rule include testing during the different manufacture stages, during the assembly of the components and after cold and hot test before start-up. The test result evaluations are used as zero-basis document for comparison with results from ISI (in-service inspection). They are an indispensable element of the quality assurance of the plant manufacture and are the basis for the evaluation of the state and integrity of the components.

The Czech experts provided an overview of NDT performed for the main primary circuit components. Deviating from European practices, no techniques were used which ensure reliable detection of severe crack-like defects in the reactor pressure vessel, especially of the most dangerous defects, i.e. cracks perpendicular to the surface (underclad cracks, near-surface cracks). Such techniques are included within ISI, and Czech experts suggested a TOFD (Time Of Flight Diffraction) technique. However, this technique is not internationally validated. The only internationally accepted techniques with the necessary capability are the Tandem technique together with a V-transfer approach and the French focussing technique. Extensive international investigations (PISC) have demonstrated that the detection probability for the above mentioned defects is very low without Tandem or V-transfer methods.

In Germany, Switzerland, the Netherlands, and the UK (Sizewell B) and for the inspection of several WWER reactors Tandem and V-Transfer techniques are integrated into the standard NDT programmes (e.g. these techniques have been used for pre-service inspection in Mochovce NPP, a WWER 440/213-type reactor).

The sensitivity of the NDT methods used in NPP Temelín and thus the crack detection probability could not be assessed because the detailed test results were not available at the NPP. There is also no summarising report on the mechanised ultrasonic tests of the reactor pressure vessel (RPV) and the primary circuit pipes, although these tests were performed more than half a year ago (before August 2000). According to European practice such reports have to be available at the plant before the first fuel loading of the reactor, and certainly before the operation permit is issued. This is also true for a summarising report on all results of NDT of the secondary circuit (feedwater piping, steam generator collector) before start-up. It should include a description of the used testing methods, sensitivities, a comparison of the testing extent required by standards and licensing requirements with the extent actually realised, and a list of all indications above the registration level. This summary of registered indications is the basis for the manufacture/assembly quality evaluation within the licensing procedure and also a basis for later ISI-comparison purposes. The Czech Regulation 436 of the CAEC ([1]) , Oct. 10, 1990, § 5 (2) and § 8 (1)/(6) also requires the safe storage of the complete documentation on design, construction, manufacture, assembly and erection quality assurance by the Operator.

The presentation on the NDT programme, extent and results by Czech experts from Skoda Plzen (manufacturer of the main components and responsible for the non-destructive testing before start-up) concluded with the statement, that no unallowable indications were found or left in the main components. However, the review of some parts of the POSAR and the reactor passport revealed at least three indications, which exceeded the size limits defined in the respective normative regulations. Detailed in-depth investigations of these indications applying additional non-destructive test procedures as is common practice in Europe were not performed. According to regulations static strength and fatigue calculations have to be performed in case of non-allowable indications in order to preclude an uncontrolled crack propagation during operation. Although requested by Austrian experts such calculations were not made available for review. The reasons given for to leaving these defects unrepaired was either that the component had already passed several tests (pressuriser) or that other (less stringent) regulations were applicable due to the in-service status reached by now (ASME Code). Possible damage by repair was also not assessed.

NDT problems also exist for the circumferential welds of the main steam lines as well as for the weld-on wedges for the pipe whip restraints. Up to now only one of the welded wedges (“fixations”) has been inspected performing an incomplete UT-inspection. The inspection is not complete, because the pipe base metal area below the fillet weld was not inspected for possible underweld-cracks (e.g. with specially qualified creeping wave probes). A specific inspection procedure is therefore recommended for the fixation wedges at the main steam line at the +28.8 meter level. Special precautions must be taken to ensure the reliability of these inspection results.

Potential impacts on Austria: The RPV and the primary circuit pressure boundary are one of the safety barriers to the radioactive core and thus to almost all of the radioactive inventory of the NPP. As there is no barrier designed to withstand the impact of a sudden RPV rupture, the structural integrity of the RPV is of utmost importance for the enclosure of the radioactive inventory. This integrity must be maintained under normal operation, anticipated transients and accident conditions. Non-detected cracks might start to grow in case of a PTS (pressurised thermal shock) transient and crack propagation can result in a catastrophic RPV failure. Non-detected cracks in the primary/secondary circuit might grow during operation, which might lead to loss of coolant accidents (LOCAs) of various sizes. Detection and monitoring of cracks is therefore an essential safety requirement.

The IAEA WWER 1000 Issue book (Issue CI2) gives such deficiencies a high ranking (RANKING III): “Some deficiencies have been revealed, related to vessel inspection from outside, testing of underclad area, and testing of steam generator collectors and tubing.”

3.5 Technical Arguments

In the following, the results of two bi- and trilateral consultations concerning the pre-service NDT performed by the Czech side shall be discussed with respect to the basic principles on NDT elaborated in the next section and in comparison with European practice. Czech co-operation during these high quality consultations prepared by competent experts is highly appreciated.

The workshop in Rez on February 26, 2001 was devoted to NDT; two presentations by the Czech experts were given:

a) Petr Tendera (SUJB): Legal basis for ISI programme and respective POSAR chapters on ISI description and review made by SUJB; Czech Presentation to the Expert Mission with Trilateral Participation on the TEMELIN NPP Unit-1; February 26, 2001 (see Attachment 1)

b) Ladislav Horacek (Nuclear Research Institute Rez): Non-destructive Testing (Integrity of Primary Loop); Czech Presentation to the Expert Mission with Trilateral Participation on the TEMELIN NPP Unit-1; February 26, 2001 (see Attachment 2)

Following the presentations numerous questions were answered and the laboratory Rez was visited.

A second workshop concerning selected questions took place at NPP Temelin on March 16, 2001.

The discussions allowed an overview concerning the application and fulfilment of the above mentioned principles. It must be stressed that this overview is not necessarily complete because due to the restricted time schedule only spot-checks of selected documents were possible.

3.5.1 Principles of non-destructive testing in nuclear power plants

The aim of NDT assessment is the quality of relevant components of the primary and secondary coolant circuit with respect to cracks and material defects that could jeopardise the strength and thus the component integrity. The main principles of NDT (non-destructive evaluation) are:

- Sufficient sensitivity: non-allowable defects that could impair the strength should be detected with a probability close to 100 %. This is especially important for cracks with flat two-dimensional geometry, and special test procedures are necessary to guarantee their exclusion with high confidence.

- Repeatability: testing methods and test technologies should be selected in view of their use during ISI (in-service inspection) with respect to accessibility and radiation exposure. This is necessary for the detection of a possible growth of defects due to operational loads.

- Accessibility of testing locations: construction and assembly of components have to allow a 100 % inspectability of the component. For complex constructions it is necessary to use specific test procedures that guarantee a 100 % volumetric inspection.

- Independence: The inspection has to be performed or supervised by independent institutions. This is requested to avoid conflict of interest with respect to the quality assessment during manufacture on the one hand and operation by the Operator on the other hand.

- Comprehensive documentation: The NDT documentation has to include the complete information concerning the testing technology, the reached sensitivities, the inspection volume and test results for all observed indications.

- Reliable evaluation criteria: for the evaluation of indications (defect characteristics, categories of defects) reliable criteria have to be defined.

3.5.2 Insufficient detectability of cracks lying perpendicular to the surface in primary circuit components

All WWER-1000 primary circuit components (reactor pressure vessel and head, primary coolant pipes, reactor pump housing, pressuriser and steam generator collector) are made from ferritic steel with austenitic cladding for corrosion protection. In these components underclad cracks can develop that will propagate from the ferritic/austenitic interface perpendicular to the surface into the ferritic material. The possibility of crack formation perpendicular to the surface in the weld metal or the heat affected zone (coarse grain region) also exists for the main circumferential welds. These defects represent the largest hazard for the strength and integrity of the component compared to other volumetric defects (pores, inclusions etc.); therefore a high detectability is required in order to secure the structural integrity. International round robin tests on test blocks with defined defects (PISC) were performed in order to characterise the efficiency of different ultrasonic test procedures with respect to the detectability of the specified defects. These investigations [2] revealed that the detection of cracks perpendicular to the surface could not be reached reliably using ultrasonic single probe tests:

“*Importance of supplementary probes: Procedures in the spirit of ASME section XI appear to perform better if a 70 % longitudinal waves angle probe is added either in the simple echo technique or with dual beam, or both. Figure 2 on plate No. 2 shows the importance of supplementary techniques: an ASME procedure at 35 % DAC complemented with 70° angle probes and tandem probes matches the detection rate achieved with the 10 % DAC procedure. Detection of near surface cracks is very important from a structural integrity viewpoint.”*

In consequence the application of specific test procedures for the detection of this type of defects was demanded:

Report 29 [3] states on page 81: "*The group of true "alternative procedures" showed a very significant improvement in performance when compared with standard PISC procedure (ASME) XI); the use of high sensitivity echo techniques using multiple orientation and tandem techniques, and the use of focusing probes gave nearly perfect detection. As a consequence of this improved detection these alternative procedures were able to reject correctly all the composite defects whereas none was rejected by the PISC (ASME XI) procedure. Two of these procedures were routinely used in Europe for the in-service inspection of reactor pressure vessels.*"

Detail requirements of the application of the Tandem technique are included in European national standards (German KTA regulations KTA 3201.3 [4], KTA 3201.4 [5]):

The German KTA 3201.4 ([5]) regulations require the use of the Tandem technique for volumetric investigations:

Lfd. Nr.	Prüfart	Prüfverfahren		Prüftechnik
1	Prüfung der Oberflächen	magnetische Streufluß-Verfahren	(MRV)	Magnetpulverprüfung (MP), Streuflußprüfung
		Eindringverfahren	(FE)	z.B. Farbeindringprüfung
		Ultraschallprüfverfahren	(US)	z.B. Oberflächenwellen, Modenumwandlung, SEL-Prüfköpfe, elektromagnetisch erzeugte Ultraschallwellen
		Wirbelstromprüfverfahren	(WS)	Einfrequenztechnik, Mehrfrequenztechnik
		Durchstrahlungsprüf-verfahren	(DS)	Röntgentechnik, Isotopentechnik
		Gezielte Sichtprüfung	(SP)	mit oder ohne optische Hilfsmittel
2	Volumenprüfung	Ultraschallprüfverfahren	(US)	z.B. Einkopftechnik mit Senkrecht- oder Schrägeinschallung, Tandemtechnik, Modenumwandlung
		Durchstrahlungsprüf-verfahren	(DS)	Röntgentechnik, Isotopentechnik
		Wirbelstromprüfverfahren für dünne Wandungen	(WS)	Einfrequenztechnik, Mehrfrequenztechnik
3	Integrale Prüfung	Integrale Sichtprüfung		—
		Druckprüfung		—
		Funktionsprüfung		—

Tabelle 2-1: Prüfarten, -verfahren und -techniken

According to KTA 3201.3 ([4]), paragraph 13.2.5.4.1 the Tandem technique is required for the inspection of ferritic I-weldings with nominal wall thickness greater than or equal to 40 mm and for all weldings with wall thickness greater than or equal to 100 mm.

During the discussion on the ultrasonic testing of the reactor pressure vessel the Czech experts stated that no specific test corresponding to the Tandem technique has been used. This is astonishing as the Tandem inspection had been applied for other Skoda projects (e.g. NPP Mochovce). The Czech experts assume that there is enough redundancy in an ISI system, which can carry out inspections from the outer and the inner side of the reactor pressure vessel wall. Such an argumentation can not be accepted because there are access limitations for the inspection for the inside as well as for the outside. Also the low detection level of the techniques employed here cannot be compensated. The PISC II study showed that only a 20 % DAC level of the ASME Code, Section XI, applied during the 1980ies may give a certain guaranty to detect even large planar cracks perpendicular to the surface.

Skoda experts gave detailed information on the detection sensitivity of the ultrasonic test methodology: The actually applied sensitivities are different for wall thickness below and above 200 mm. Three amplitude thresholds for “Recording”, “Registration” and “Acceptance” are used for all single element probes (0, 35, 45, 60, 70):

- for wall thickness below 200 mm (in terms of DGS-diagrams equivalent flaw sizes/mm): Rec: 2.8; Reg: 3.8; Acc: 4.5 and
- for wall thickness above 200 mm: Rec: 2.8; Reg: 5.1; Acc: 6.2.

For the TRL-Probe TRL 70 a flat bottom hole of 4 mm diameter perpendicular to the cladding interface in the base material is used. As a substitute for the Tandem technique through possible redundancies attained by internal and external inspections a special validation programme is under development at Skoda (executed probably together with the NRI). This programme will also include the TOFD-technique (Time Of Flight Diffraction) for the weld root area. So far, the TOFD-technique is not internationally validated. It is at present not possible to assess the potential of this project.

The NDT results of the RPV are often used within the structural integrity assessment. Several code regulations assume that using specific test procedures, large cracks can be excluded so that for a PTS (pressurised thermal shock) analysis esp. in case of the development of cold plumes it is no more necessary to postulate semi-elliptical cracks up to the ¼ wall thickness. In the case of NPP Temelín this assumption is not valid because of the low detectability of cracks lying perpendicular to the surface.

For the other cladded primary circuit components there was apparently no special ultrasonic technique used for the detection of underclad cracks or other cracks perpendicular to the surface. The applied techniques may offer sufficient potential for this problem, but this can not be assessed for the time being, because the qualification project for those components has not yet been carried out at the NRI. The Czech NDT technology is qualified for stainless steel primary piping, but in contrast to the WWER-440 project, where these components are made of austenitic steel, for the WWER 1000 the primary coolant piping, the steam generator collector and the reactor pump housing are manufactured from ferritic steel.

<u>The conclusions</u> with respect to the detectability of cracks lying perpendicular to the surface are as follows:

- Deviating from European practice (including other WWER reactors) no Tandem or French Focussing technique was used. Thus the detectability of cracks lying perpendicular to the surface is too low.
- An ultrasonic testing technology with sufficiently high detectability of cracks lying perpendicular to the surface needs to be implemented.
- Until state-of-the-art methods have been applied it is not acceptable to take credit for the performed NDT programme to reduce the postulated crack size for the RPV structural integrity assessment.
- The NDT programme for the other primary circuit components has to be qualified with respect to cracks lying perpendicular to the surface. The test procedure needs then to be performed at the earliest possible time.

- For Temelín NPP Unit-2 there is still time to implement the NDT programme in accordance with international practice as part of the pre-service inspection, including the Tandem technique or an equivalent method with sufficient detectability of cracks lying perpendicular to the surface.

3.5.3 NDT qualification programme

A number of test blocks for the primary coolant circuit of the WWER 1000 reactor is available at Rez, but apparently the programmes for the qualification of NDT for some more unusual and complicated geometries of this type of pipes has not been started.

Mr. Horacek reported on a validation project supported by the PHARE programme, which gave the frame for the qualification of some ISI techniques for pipes and special NDT-problems typical to the primary coolant loop. During the presentation of Mr. Horacek it became clear that this qualification programme was especially related to problems of the WWER 440 reactors with stainless steel primary circuit pipes. The WWER 1000 contains essentially ferritic primary coolant pipes with an austenitic cladding at the inside. That means that most of the non standard weld situations of this type of cooling circuit cannot be regarded as qualified by the previous PHARE activities. It has therefore been pointed out that for more complicated geometries (e.g. nozzles and dissimilar metal welds) one has to consider a separate programme because the experiences from the WWER 440 circuit cannot be transferred to the WWER 1000.

The German regulations KTA 3201.4 ([5]) require test blocks made from materials identical to the component, with identical acoustic properties and representative geometry[2].

The common position of European Regulators on qualification of NDT systems for pre- and in-service inspection of LWR components [6] states:

> 5.3: „A practical assessment is a performance demonstration of a NDT system using qualification test blocks representing the component or part of the component (its geometry, material composition, material structure, fabrication and surface conditions) and the defective conditions the NDT system is intended to detect and sentence. The representativeness of defects in the qualification test blocks is a key point in any practical assessment ...“.
>
> 7.1: Responsibilities for NDT qualification activities in any European country must be consistent with its legal system and regulatory practices.

[2] KTA 3201.4
4.2.3.3
(4) Kontrollkörper müssen so beschaffen sein, daß eine reproduzierbare Einstellung der Prüfempfindlichkeit sichergestellt ist. Kontrollkörper sollen stets unplattiert und aus Werkstoffen hergestellt sein, welche die gleichen akustischen Eigenschaften wie die Werkstoffe der Prüfgegenstände aufweisen.
(5) Bei akustisch schwierig zu prüfenden Werkstoffen und geometrisch komplizierten Konturen sind Vergleichskörpermessungen durchzuführen und die sich hieraus ergebenden Transferkorrekturen für die Prüfempfindlichkeit zu berücksichtigen.
(6) Vergleichskörper müssen in Geometrie und akustischen Eigenschaften repräsentativ für die zu prüfende Komponente sein. Wenn die Gegenoberfläche bei der angewendeten Prüftechnik wirksam ist, sollten Abweichungen von der Wanddicke des zu prüfenden Bauteils kleiner als 10 % dieser Dicke sein.
(7) Die in Vergleichskörpern eingebrachten Reflektoren müssen in Anzahl und Variation der Abmessung und Lage ausreichend sein, um Aussagen zum Nachweisvermögen der Prüftechnik zu ermöglichen. Sichtprüfung mit dem menschlichen Auge und mit einem Gerätesystem, das die Bildinformation aufnimmt, weiterleitet, darstellt oder speichert.“

Conclusion: The validation and verification of the NDT project for the primary circuit components (especially with respect to ferritic tubes with internal stainless cladding) should be accomplished as soon as possible to make quantitative analyses of pre-service inspection results available.

3.5.4 Insufficient documentation of test results

The NDT result documentation is part of the general documentation of an NPP, as required in numerous national and international regulations. The NDT documentation is also part of licensing documentation and has to be archived accordingly.

As an example the respective German regulations are cited:

KTA 1401 ([7]) general requirements for quality assurance, 7. manufacture, assembly, erection including quality assessment, 7.2 performance and surveillance of manufacture, assembly, erection and inspection (5): the inspection according paragraph 2 and the surveillance according paragraph 4 have to be documented pursuant the process of work. The results can therefore be evaluated in time for immediate introduction of corrective measures[3]. 8. Start-up (2) The documentation for start-up has to include all important information corresponding to safety relevant requirement.

KTA 1404 ([8]) documentation during manufacture and operation of nuclear power plants, 3. General requirements on documentation, 3.3 quality documentation (5): It has to be secured that the required documentation about the manufacture is performed, summarised and reviewed. [4]

KTA 3201.3 ([4]) Primary circuit components of light water reactors. Part 3: manufacture, 4.2.4 Documentation at the licensee (final documentation deposit E)

(3) The final documentation has to include component related: c) the pre-inspection documents characterised by E and the results of manufacture and partial manufacture inspections; 2.22 ultrasonic testing c) Tandem volumetric testing H, S E,

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[3] KTA 1401 Allgemeine Forderungen an die Qualitätssicherung
7. Fertigung, Montage, Errichtung einschließlich Qualitätsprüfungen
7.2. Durchführung und Überwachung von Fertigung, Montage, Errichtung und Prüfungen
(5) Die Prüfungen nach Absatz 2 und die Überwachung nach Absatz 4 sind dem Stand des Arbeitsfortschritts gemäß mit Unterlagen zu belegen. Die Ergebnisse der Prüfungen sind so rechtzeitig zu beurteilen, daß noch Korrekturmaßnahmen am Produkt in die Wege geleitet werden können.
8. Inbetriebsetzung
(2) Die Inbetriebsetzungsunterlagen müssen entsprechend den sicherheitstechnischen Erfordernissen alle wesentlichen Angaben für die Inbetriebsetzung enthalten. Hierzu gehören:
a) das Ziel des Inbetriebsetzungsvorgangs,
b) die Zustände der benötigten Systeme,
c) die Handlungen zum Erreichen der Zustände,
d) die jeweils zu beachtenden Grenzwerte,
e) Angaben über erforderliche Protokollierungen und zu archivierende Prüfprotokolle und Prüfgrundlagen (Inbetriebsetzungsdokumentation).

[4] KTA 1404 Dokumentation beim Bau und Betrieb von Kernkraftwerken
3 Allgemeine Forderungen an die Dokumentation
3.3 Qualitätsdokumentation
(5) Es ist sicherzustellen, daß alle für die Dokumentation erforderlichen Unterlagen herstellungsbegleitend erstellt, zusammengestellt und geprüft werden.

(4): All manufacture and assembly documents designated for the final documentation must be available, evaluated and verified system related at the time of system pressure test.
(6) The documents of the final documentation have to be stored by the licensee. [5]
The Czech Regulation 436 of the CAEC, Oct. 10, 1990, on quality assurance of classified items from the point of view of nuclear safety of nuclear power plants states in § 5 (2) that the organisation that operates the NPP has to have an overview on the technical state of the constructed and operated plant in order to be able to demonstrate that the projected quality was reached and assured during documented design, manufacture, assembly and erection. And in § 8 (1): All activities that affect the component quality, esp. during design, construction, manufacture, storage, transport, assembly, erection, inspections and tests, start-up, maintenance, outages, repairs and upgrading have to be performed according to an approved documentation. This documentation is part of the quality assurance and has to include the procedures and methodologies of the quality control and the acceptance criteria. § 8 (6): The documentation about the activities that affect the quality of the specific components has to be stored by the Operator of the plant during the complete operating time of the plant at two different, with respect to safety, independent sites.

The discussion with the Czech side revealed that the NDT documentation is incomplete and partially not available. At the time of the last workshop in March 2001 no final report on the mechanised ultrasonic testing of the reactor pressure vessel and the mechanised ultrasonic testing of the primary coolant pipes was available, although these pre-service tests were performed before loading of the reactor core in the middle of the year 2000. Merely some individual protocols with strongly compressed information content were found in the component passports. A review of the test plots, for instance for the reactor pressure vessel, in order to evaluate the reached sensitivity of the ultrasonic test procedures was not possible, because this documentation was not at the power plant.

There was no summarising overview available on the pre-service NDT of the safety relevant components of the primary and secondary coolant circuit. Such an overview must include a description of the volume and content of the legally required inspections, the volume and content of the actually performed inspections, the testing technology used and the respective sensitivities as well as a list of all observed non-allowable indications (indications above registration limit). The presentation by Mr. Horacek cannot replace this requirement because of its incomplete information. The presentation contains the conclusion „All indications found within the pre-service inspections have been assessed and considered to be allowable in compliance with the acceptance criteria of appropriate standards." This statement could not be confirmed by the Austrian experts because such a list of indications above registration limit does not seem to exist. On the other hand a very short spot-checking of the component documentation revealed that at least three non-allowable indications were found during preservice NDT (see also next chapter). The insight into the component documentation was strongly restricted due to condensed time schedule and language problems. Therefore a strong uncertainty remains regarding the question whether all non-allowable defects with potential hazards for strength and integrity of the main

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[5] KTA 3201.3, Komponenten des Primärkreises von Leichtwasserreaktoren, Teil 3: Herstellung
4.2.4 Dokumentation beim Genehmigungsinhaber (Endablage E), (3) Zur Endablage gehören komponentenbezogen: c) die in den Vorprüfunterlagen mit E gekennzeichneten Nachweise und Ergebnisse von Bau- und Teilbauprüfungen, Prüfungen und Nachweise, 2.22 Ultraschallprüfungen, c) Tandem-Volumenprüfung H, S E, (4) Alle für die Endablage bestimmten Unterlagen der Herstellung und der Montagearbeiten müssen systembezogen bis zum Zeitpunkt der System-Druckprüfung geprüft vorliegen und zusammengestellt sein.
(6) Die Unterlagen der Endablage sind beim Antragsteller oder Genehmigungsinhaber sachgemäß aufzubewahren.

components were registered, analysed and evaluated with respect to their importance. Thus in the view of the Austrian experts the quality assessment with respect to defect-free manufacture and assembly of the main components was not demonstrated.

Conclusions on the NDT documentation:

- No final documentation of the mechanised ultrasonic testing of the reactor pressure vessel and the primary circuit is available.

- No summarising evaluation on the pre-service NDT inspection of the safety relevant main primary and secondary circuit components including the legally required and actually performed NDT programme, the test technologies and detection sensitivities, and an overview on all indications above the registration limit and their evaluation was available. There is no statement regarding the grounds that permitted the Regulator to license the plant to the present extent in spite of this obvious violation of Czech regulations.

- The missing documents need to be worked out in the short term.

3.5.5 Non-allowable indications

All codes require that indications observed during NDT-pre-service inspection above a defined recording limit have to be classified, generally within two categories: indications above the registration limit and indications above the acceptance limit (“non-allowable indications”). These non-allowable indications have to be analysed further and their origin has to be clarified in order to decide upon their final acceptance based on more reliable knowledge. In case of insufficient clarification they have to be controlled periodically during in-service inspections (ISI) in order to detect possible operationally induced development or growth (due to mechanical or corrosive effects). If a critical assessment shows that indications exceed the acceptance limit the respective measures (repair or component replacement) have to be taken.

Only in exceptional situations individual non-allowable indications might be left unrepaired. In such cases special test procedures including in-depth testing methods and combinations of methods have to be performed in order to describe location and geometry very precisely. Furthermore, a conservative classification of the indication, usually as crack with enveloping geometric contours, has to be assumed. Closely lying indications have to be evaluated as total defect with respect to their safety relevance. This evaluation includes the analysis of the actual degradation of the component stability taking into account all relevant loads from normal operation, hydrotests and accident situations. Usually linear elastic fracture mechanical methods (brittle fracture resistance) or elasto-plastic fracture mechanics (stability and integrity assessment), demonstration of leak-before-break (LBB) behaviour are applied. Besides the observed defect geometry the growth of the defect due to mechanical and/or corrosive loads, and changing material properties (neutron embrittlement, low-cycle fatigue, thermal ageing) have to be taken into account. It is necessary to use conservative assumptions in order to ascertain that the defect is not developing into a strong hazard for the strength and integrity of the respective component. The results of this evaluation in case of an unrepaired non-allowable indication have to be forwarded to the licensing authorities, who have to participate in the relevant decisions.

This procedure is international practice and can be found in the respective regulations. This procedure is also applied in Russian-type reactor components of nuclear power plants in Eastern Europe. The normative basis during manufacture is PK 1514 dating from 1972 (this regulation is no longer valid in Russia). The PK 1514 were supplemented by national regulations. This step is not yet completed in the Czech Republic. In 1998 a draft for “Guidelines and recommendations for the evaluation of the reactor pressure vessel and the reactor internals of WWER NPP during the operation of the NPP” was published by the SUJB. This document is restricted to the reactor pressure vessel and the reactor internals, no other main components of the primary and secondary side are included. As of today, there is still no final version and no agreement on the obligatory application of the draft. The introduction to this document states:

> “*The ‘Guidelines and recommendations....’ are supposed to be guidelines for the licensee for the preparation of a safety documentation according to the attachment of the law No. 18/1997 Slg., part E and F, for the lifetime assessment (integrity evaluation) of the reactor pressure vessel and the reactor internals. The draft of the ‘Guidelines and recommendations....’ will be the basis for the final version, that will be edited by SÚJB as ‘Safety Guidelines’ on the basis of objections and comments. If the licensee will be using the final version of ‘Guidelines and recommendations for the evaluation of the reactor pressure vessel and the reactor internals of WWER NPP during the operation of the NPP’ after it’s edition the respective part of the safety documentation will be accepted as adequately fulfilling of the legal requirements*.”

During the workshop on February 26, 2001 there was no agreement among the Czech experts about the validity of the PK1514 [9] and the ‘Guidelines and recommendations ...’. According to Mr. Brumovsky the application of PK1514 [9] is restricted to the inspections at the manufacturer and after the final assembly of the plant, while the NDT inspections during pressure testing in the commissioning phase are performed according to the ‘Guidelines and recommendations ...’. The representative of the licensing authority, Mr. Tendera, stated that ‘Guidelines and recommendations ...’ will only be applicable for in-service inspections after commissioning.

In any case, both regulations include the same methodology in case of indications, in compliance with international practice. The attachment 3 of ‘Guidelines and recommendations ...’, paragraph 1.2 states:

> “*The methodology has to include the following steps:*
>
> 1. *Categorisation of the observed defect in a way that can be used for calculations*
> 2. *Comparison of the schematised defect with the tabulated values of acceptable defect sizes in case, that the schematised defect is smaller than the acceptable sizes in table 3.1. to 3.4 the defect is allowable. In case that the schematised defect is larger than the acceptance limit in the respective tables 3.1 to 3.4 the following procedures have to be performed:*
> 3. *Determination of the temperature field at the wall of the pressure vessel including cladding*
> 4. *Determination of the stress field in the environment of the schematised defect*
> 5. *Determination of the stress intensity factor K_I (or J) for the schematised defect*
> 6. *Selection of the allowable values of fracture toughness K_{ic} (or J_{ic} or $J_{0.2}$)*
> 7. *Nature of influence of degradation mechanisms*
> 8. *Estimation/ accounting of the possible safety margins*

9. *Calculation of the effect of the schematised defect on the lifetime of the pressure vessel*
10. *Decision on the acceptability of the defect.*"

Summarising, it can be stated that the normative regulations for the required evaluation procedures in case of non-allowable indications is basically identical within the different codes, but the old codes (like PK1514/72 [9]) do not reflect the progress of fracture mechanical methods for the strength evaluation of components containing defects.

Therefore the handling of non-allowable indications in the three uncovered cases of large defects found during pre-service inspection is quite astonishing. In all three cases the defects are non-allowable according to PK1514 ([9]) and also according to the Czech 'Guidelines and recommendations ...'. These three non-allowable indications were "found" by the Austrian experts during a very brief review of just two documents (reactor passport and POSAR), as a summarising list of preservice inspection results is still not available (see above). While these three indications will be discussed in more detail, it has to be emphasised that the Austrian experts are not sure that these three non-allowable indications are the only detected indications above acceptance limit.

The following indications were categorised as non-allowable:

- Reactor pressure vessel: weld no. 2, testing protocol JAD88-PP/118/00: defect of type B No. 3, in the weld center (y=1mm, Z=51mm from the inside); DN=5.0 mm: point defect in the area of the weld root – probably a non-metallic inclusion (produced by a 0° L-wave ultrasonic probe with an echo amplitude more than 6 dB above the recording threshold and according to the rules to be applied: non-allowable).

- Reactor pressure vessel: cladding, lower head, testing protocol JAD88-PP/117/00: defect of type D: binding defect or slag at the interface pressure vessel and cladding, laminar indication: l_x=5-55 mm, l_y=4-65 mm. Most indications are smaller than 1250 mm^2 (maximum allowable size according to Czech regulations), except indication 23 with a size of 2808 mm^2.

- Pressuriser: upper circumferential welding, non-conformance document no. 41, defect indication from ultrasonic testing with equivalent reflector size 6.7 mm.

None of the three indications was repaired. There is no information on special in-depth investigations, should they have been made.

Subsequent to the mechanised inspection the **indication in weld no. 2** was tested using manual ultrasonic methods, and "after the second inspection the amplitude could be reported to be a little bit less than the recording level". This was reported to be the formal reason to classify this indication as acceptable. But the acceptability of such an indication (probably due to some slag inclusion or an oxide skin between welding passes) should be based on more meaningful analysis (e.g. comparison of the indications produced by different angles, from different positions, synthetic aperture focusing analysis for the estimation of depth extension and others) and an attempt to ascertain the most probable origin and extension of this indication should be made. Based on more detailed information a critical engineering assessment may end up with the conclusion that this indication is not safety relevant, but simply "playing around" with the amplitude is a poor argumentation.

It is necessary to remark that the indication in weld no. 2 was detected for the first time during the mechanised ultrasonic testing after the hot hydrotest, although according to the information of the Czech experts the manufacture processes were accompanied by detailed non-destructive testing during welding processes, weld inspection after completion, after cladding, after heat treatments and after the completion of the reactor pressure vessel and the strength pressure test at the manufacturer. This interesting fact was not evaluated.

Principally two interpretations are possible:

1. The sensitivities of the NDT methods applied during manufacture were not sufficient to allow the reliable detection of this defect.

2. During the cold and hot tests of the primary circuit in the pre-service stage a defect development has occurred (opening of a material inhomogeneity that was not detected up to then, and/or growth of the defect due to the pressure loading and the first temperature gradient load during hot testing).

Both interpretations carry implications relevant for RPV integrity that need to be assessed.

For the **indication in the RPV lower head** a detailed analysis under conservative assumptions has not been performed. The defect was simply classified to be a lack of fusion between the cladding and the base material. No surface inspection of the inner cladding surface was performed in order to confirm that the cladding area of concern is essentially crack-free and is not totally separated from the base material within that area. Ultrasonic testing from the outside would have been needed to assure the existence of a sufficiently intact ligament structure between the indicated potentially week spots. Although the indication from the mechanised ultrasonic testing was larger than double the limit, the defect was not repaired. Instead, reference was made to other regulations (ASME Code), purportedly valid at this stage of operation.

Here, the same comments apply as for the indication in weld no. 2: This large defect in the lower head was detected for the first time during the mechanised ultrasonic testing after the hot test. No surface crack testing of the cladding from the inside was performed after UT (ultrasonic testing). There is no information on dye penetration testing at an earlier stage. There is information in the reactor passport that after completion of the lower head cladding repairs were performed due to quality deficiencies. It was not possible to clarify if the location of these repairs correlates with the location of the UT indications.

After the welding process and the heat treatments the interface between the cladding and the base material is exposed to high stresses during the hot hydrotest resulting from the different thermal expansion coefficients and the loads from inner pressure. This load combination occurs for the first time during the hot hydrotest. As it is highly unlikely that such a large defect is not discovered during the different manufacturing stages, it might follow that this defect grew to the reported geometric extensions. Therefore it is highly unusual that no repair was performed or at least an additional surface crack test was used to exclude cracks at the cladding surface.

The only reason given for not repairing the non-allowable **indication in the upper weld of the pressuriser** was that the manufacturing of the component including cladding and heat treatment and the pressure tests were already completed.

In all three cases the national regulations and the international practice for handling non-allowable indications were violated, since no strength evaluations of the defect containing components were performed. This had occurred under the supervision of the respective regulatory authority that has apparently accepted the procedure without accounting for this decision.

The use of ASME Code regulations for the classification of the non-allowable indication in the reactor pressure vessel has to be discussed very critically: National code regulations are always the result of several different components reflecting the specifics like national material characteristics and manufacture technologies, design parameters, strength calculations and the often very different plant/project specific load conditions. They are tuned to each other and represent an integrated package. The extraction of one specific criterion and its application on alien components needs an extensive reasoning and a specific legitimisation by the regulatory authority. Otherwise the suspicion might arise that different codes are used as needed to avoid a possibly costly conservative evaluation.

Conclusions:

- It is evident that three non-allowable indications (two in the reactor pressure vessel, one in the pressuriser) were left unrepaired without sufficient foundation. The in-depth investigation of these indications was insufficient, no strength assessment was performed.
- These defects have to be re-assessed immediately using strength calculations.
- A defect specific additional test programme (in-depth defectoscopy) has to be realised at the next possible date.
- The validity of the 'Guidelines and recommendations ...' and the PK1514/72 ([9]) has to be defined in a definite way by the regulatory body. The handling of non-allowable indications in the reactor pressure vessel and other components has to be regulated.

3.5.6 Main steam line welds at the +28.8 m level, pipe whip restraint welded wedges

During the workshop at the NPP Temelin on March 16, 2001 the Austrian experts received the information that the circumferential welds of the main steam lines on the +28.8 m level were only tested using X-ray radiography, the wedges for the pipe whip restraints welded on the main steam lines were not inspected after manufacture (only an acceptance test weld was inspected), ISI was only performed on one wedge.

The Czech side claims that X-radiography after manufacture can exclude the existence of hazardous cracks. However, experience has shown that cracks aligned to the X-radiation cannot be detected and is therefore insufficient[6].

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[6] Common position of European regulators on qualification of NDT systems for pre- and in-service inspection of LWR components [Rev. 1, 1997, EUR 16802 EN, Appendix 4]:

The ASME XI Appendix VIII provides minimum requirements, thought at a more specific level than this document. The actual implementation in the USA of ASME XI, Appendix VIII may exceed the ASME Code minimum requirements. Utilities in the USA have joined together in an organisation to implement the requirements of ASME XI, App. VIII. The following aspects of Section XI concerning ultrasonic examinations must be emphasised:

- IWA-2230 defines volumetric examination and three methods of volumetric examination (radiographic, ultrasonic, eddy current); IWB-2500 and within table IWB-2500-1 requires volumetric examination in certain cases for class-1 components; similar requirements are in the corresponding subsections for other classes.

Taking into account that as a rule any welded attachments on steam lines should be avoided it is not understandable that no ultrasonic testing was performed after the weldment. The testing of an acceptance test piece cannot replace the inspection of the real pieces after manufacture.

Furthermore there is no justification for not including in the planned ISI (in-service inspection) special procedures to detect cracks below the wedges in the pipe material (e.g. with specially qualified creeping wave probes) which have to be considered to be most hazardous defects.

It must also be kept in mind that - deviating from German or French practice - the main steam lines and feed water lines are not physically separated, therefore great efforts must be made to avoid any possibility of pipe failure.

Conclusions: A mechanised UT procedure needs to be included in the ISI programme for the circumferential welds of the main steam lines. Complete ultrasonic testing including special test procedures with respect to possible underweld cracks at the welded wedges for pipe whip restraint needs to be performed immediately.

3.5.7 Leak-before-break (LBB)

The Austrian concerns regarding LBB were not eliminated: The Austrian experts had no opportunity to review and evaluate additional documents besides the POSAR. The so called “Handbook LBB” (cited in [3.6.3-43] J. Rjdlovy, P. Samohil: Teoreticky manual k Handbooku LBB UJV 9925, [3.6.3-49] P. Samohjl, F. Kaspar: Hodnoceni primorniho potrubi JE Temelín Handbook LBB, UJV 10 469 T) was never provided to Austrian experts, however a very early Czech LBB write-up document was available. There is no information on the LBB concept[7] as approved by SUJB for the NPP Temelin.

In case the results of Czech LBB calculations are applicable to the extent required, the compliance demonstrated by fracture mechanics analyses and test programmes seems acceptable.

There is no information on the allowable leak rate detected by the leak detection systems during operation. The LBB concept is only applicable in connection with a reliable and qualified NDT programme (equipment integrity verification using appropriate ISI methods, inspection volume and intervals, and appropriate evaluation for changes with respect to material properties and usage). Since no information on the SUJB approval was presented and no adequate documentation was available it is not certain that the LBB concept realisation is acceptable and adequate.

There is also concern about the credit taken from the LBB concept implementation as a means to justify accident analyses load combinations which deviate from presumably valid code requirements in the case of SSE (safe shut down earthquake) / DBE (design basis earthquake) events: For example the load combination DB (double ended break) + EZ (earthquake load/Czech version) is not assumed, at least not as coinciding events. It could

No part of ASME Section XI requires specific volumetric examination to be by ultrasonic NDT methods; however, in order to make a volumetric examination credible against the ASME Section XI defect acceptance criteria, it may be necessary to use ultrasonic examination

[7] The systems analyzed and to be covered by the LBB concept implementation are understood to be the following: main recirculation lines, pressuriser surge line, havary systems TQ piping (partly), “Russian” safety systems (SAOZ) piping.

not be verified as to whether DB is considered at least as EZ consequence in those cases when margins are exempted. (e.g. pressuriser surge line).

For those cases where applicability of the LBB fracture mechanics requirements could not be demonstrated, proof of the limitation of damage and of the absence of a secondary failure threat to equipment relevant to safety was attempted. Whether these requirements are met has not yet been clarified and needs further discussion.

Conclusions: Based on the provided documents it could not be clarified if the LBB concept was approved by SUJB. There is also no information on the allowable leak rate detected by the leak detection systems during operation. The LBB concept is only applicable in connection with a reliable and qualified NDT programme (equipment integrity verification using appropriate ISI methods, inspection volume and intervals, and appropriate evaluation for changes with respect to material properties and usage). Since no information on the SUJB approval was presented and no adequate documentation was available it is not sure that the LBB concepts realisation is acceptable and adequate.

3.5.8 Safety Culture

The NDT programme is not in agreement with general European practice in essential points. Binding Czech regulations are still missing in some instances, and in some respects the handling of the programmes does not comply with the Czech regulations. There was no indication in the documentation made available or in the presentations of more in depth evaluation required by the Regulator before accepting the deviations. There was also no indication that the first observation of non-allowable defects after the hot hydrotest was discussed with respect to the reasons (possible growth of defects during the test). Aside from the possible safety risks incurred through incomplete NDT, this indicates that there is reason for concern regarding the safety culture during the licensing process.

3.6 References

[1] Regulation No. 436 of the CAEC, Oct. 10, 1990, on quality assurance of classified items from the point of view of nuclear safety of nuclear power plants
[2] S. Crutzen, H. Herkenrath. K. Kussmaul, U. Mletzko, PISC full scale reactor pressure vessel validation of non-destructive examination, 14. MPA Seminar, 1988, 26.1-26.28
[3]: PISC Report 29, 1993, EUR 15334 EN
[4] KTA 3201.3: Komponenten des Primärkreises v. Leichtwasserreaktoren; Teil 3: Herstellung
[5] KTA 3201.4: Komponenten des Primärkreises v. Leichtwasserreaktoren; Teil 4: Wiederkehrende Prüfungen und Betriebsüberwachung
[6] Common position of European Regulators on qualification of NDT systems for pre- and in-service inspection of LWR components ,Rev. 1, 1997, EUR 16802 EN
[7] KTA 1401: Allgemeine Forderungen an die Qualitätssicherung
[8] KTA 1404: Dokumentation beim Bau und Betrieb von Kernkraftwerken
[9] PK1514-72 Regulations for welding joints and melting connections and structures of NPPs, experimental and test reactors, Metalurgia Moscow, 1974

4 Environmental and Seismic Qualification of Equipment - Issue 19

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4.1 Introduction

Environmental and dynamic qualification of equipment important to safety in nuclear power plants ensures its capability to perform required safety functions under postulated accident conditions including harsh environment (e.g. loss of coolant accidents (LOCA), high energy line break (HELB), seismic and other dynamic conditions).

Qualification of equipment is one method of preventing environmentally induced common cause damage to redundant systems and therefore to prevent the loss of safety functions.

Documentation providing evidence of the established qualification needs to be available in an accessible and auditable form at the utility, and preferably at the power plant. It is not adequate to have the documentation available at the component supplier. Records demonstrating that equipment qualification (EQ) has been established should contain information on specific equipment items being qualified, the demonstrated safety functions, applicable service conditions, qualification methods, results, limitations, justifications and relevant support technical data. These records need to be organised in an understandable and traceable manner [7, 8].

4.2 Identified Problems

In Temelín NPP, at present in the nuclear commissioning phase, qualification of ‘’safety‘‘ and ‘’safety related’’ equipment is not fully established. This is not in agreement with internationally accepted safety requirements [5, 6, 7, 8] by which the qualification of equipment important for safety has to be ensured and demonstrated before a license will be granted.

The formal qualification reports, which should have been issued, reviewed, approved and available on site before fuel loading, are not yet fully issued.

The original approach for qualification of equipment was in agreement with the original soviet design and practice, however these are not in accordance with international practice as stated also in the EGP support design documentation (see [4] EGP documentation for step 1° complex programme for EQ).

Therefore the Complex Programme for environmental re-qualification of equipment and components important to safety was developed and - according to POSAR ([9] p. 3.11.3) – implementation began in Spring 2000 and should be completed by the middle of 2002.

Three major steps characterise this programme [2] [3]:

- reassess the qualification status of safety and safety related equipment,
- take the necessary actions based on findings, and
- issue the required qualification reports and preservation programmes.

According to presentations made in the Prague trilateral meeting in March 2001 on the current status of the Complex Programme, Environmental qualification is still to be completed while seismic qualification is completed and ''archived''. It was also reported that the documentation is still under finalisation. Unfortunately, there was no opportunity to clarify some of the questions that arose from the presentations (see Attachment 1).

In addition, the presentation in Prague mainly gave information about the qualification of safety systems. It has been reported that a decision from SUJB gives priority to the process of environmental qualification for <u>safety systems</u> and the same process for <u>safety related systems</u> will be continued subsequently.

Information about the current situation of qualification of <u>safety related systems</u> and the planning of the process for full establishment of their environmental qualification was not clarified in the presentation in Prague.

No information was given about the justifications brought forward by the Operator to receive authorisation from SUJB for fuel loading without having fully established equipment qualification. Nor was there an indication of the basis and conditions for SUJB to give authorisation to load fuel and to start nuclear commissioning.

During the presentations in the Prague March meeting there was no evidence of the steps taken by SUJB in the current licensing process of the Complex Programme for qualification, its technical requirements, documentation requirements and evaluation of the results.

There are also questions as to how the Operator is managing the Complex Program for resolution of the issue. According to the POSAR [9] it seems that the responsibility to demonstrate equipment qualification in front of SUJB has been given to a contractor. This would be extremely unusual, if it means that also the primary responsibility for qualification is delegated to this contractor: a contractor can perform tests and analyses but the responsibility to ensure the qualification of equipment operating in the plant remains with the Operator. This means that the Operator has to demonstrate his own culture and knowledge in the matter of qualification, manage all the aspects related to equipment qualification, review available documentation for identification of additional actions to be taken, integrate the work performed by different contractors ensuring correct interfaces and demonstrate the correct and complete implementation of this safety requirement to the Regulator.

4.3 Solution to Identified Problems

To gain better understanding and evidence of the way this important issue has been managed and is managed it is considered essential to have:

- access to some of the seismic and environmental qualification reports onsite,
- a specific meeting with CEZ on the aspects related to technical basis and management of this issue, on the findings from the Complex Programme in progress and on related licensing aspects (including management and documentation),
- a specific meeting with SUJB on the basis and position taken by the Regulator on this item.

Neither for Temelin Unit 1 nor Unit 2 would European state-of-the-art practice permit operation or even fuel loading before resolution of the EQ issue. In any case, the completed set of the qualification reports has to be issued, reviewed, approved (also from the regulatory side) and be made available on site for Unit 1 by mid of 2002 (as stated in the POSAR [9] Rev. 1, pg. 3.11.3), and before fuel loading for Unit 2.

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

The requirement for environmental and dynamic qualification is valid in all member states, although with some differences in the practices concerning ageing, irradiation degradation models, consideration of special stresses during operation or during man-made hazards and steps of the test sequence.

The qualification of equipment important to safety has to be ensured and demonstrated before fuel loading and the formal qualification reports should be issued, approved by the Regulator and made available onsite before nuclear tests. The Operator is responsible for the overall plant specific qualification programme, which will cover all aspects of the EQ process to demonstrate fulfilment of the qualification requirements and to preserve the qualified status of equipment important for safety.

The Operator co-ordinates the support by suppliers and other technical support organisations to implement the qualification activity and submits the required relevant documentation to the Regulator endorsing the related responsibilities to ensure fully implemented qualification.

In this regard there is a note in the POSAR ([9] par. 3.11) stating that the contractor, who will be in charge of implementing this Complex Programme for re-qualification, will endorse the responsibility to defend the results in front of the Czech Regulator. Does this mean that the Operator has delegated to this contractor the primary responsibility to ensure in front of SUJB the equipment qualification in Temelin? In no way could the primary responsibility for qualification be delegated to a contractor by the Operator in the member states of the EU.

During the licensing process, the role of the regulatory authority is to verify that the licensee's EQ programme meets the applicable regulatory requirements and standards and will, as appropriate, perform audits and surveillance of implementation of qualification programme and reviews qualification reports.

In Temelín the full implementation and demonstration of EQ is not yet achieved even though the fuel was loaded in the year 2000 and the commissioning phase is in progress with nuclear tests.

In addition the required documentation showing the qualification of the equipment is still a pending issue, for which there has been no evidence, in terms of quantity and quality, of what is missing and what is available and to which extent the available documentation is ''generic'' or ''specific'' for Temelín NPP.

The issue is of significance in several ways:

- Apparently equipment of safety systems and safety related systems has been installed in the Temelín NPP being not fully qualified or at least having not the required demonstration of their capability to perform the safety functions in line with the required safety requirements under the envisaged abnormal conditions.

- The fact that commissioning and nuclear tests are ongoing in spite of incomplete equipment qualification raised concern regarding

  - the current capability of some equipment to operate in abnormal condition
  - the quality of the licensing process and the related safety culture. This in particular because of the absence of information about the basis and conditions for the decision.

- The lack of documentation on equipment qualification on site deprives the Operator of important information in case of abnormal and emergency conditions.

4.5 Comments on Czech reaction and EC comments reported in the EC paper summarising the outcome of meetings held under Melk protocol

Czech reaction

''The safety authority was prepared to explain in detail the qualification methodologies but not to allow any third country to supervise the work done by the national authority. This being said, discussion may continue bilaterally to address the issue.''

Austrian experts would be pleased to have a discussion with SUJB on the basis and positions taken in the licensing process of equipment qualification and related requirements. The intent is not to supervise SUJB work but to have direct evidence of the past and current situation and in serving this purpose SUJB should feel free to show its work.

EC comments

Enough details were given to satisfy EC experts that seismic EMC, and environmental qualification was being treated professionally. All qualifications are effectively completed. The documents are archived. Nevertheless, in some cases the documentation is being reorganised and amended in order to comply with latest requirements. (See [10] Working Paper Summarising The Outcome Of The Expert Mission With Trilateral Participation Established Under The Melk Protocol, Chapter IV).

The statement that ''all qualification are effectively completed'' is an optimistic view from EC experts because the Czech presentation refers to first phase of Complex Program related to Safety Systems.

In a second phase, to be started if understood well, they should implement the same programme for safety related systems. In addition for Safety Systems it is reported that the ''EQ process is performed'' but ''review of the EQ process'' is not yet completed. These statements do not allow Austrian experts to convene that all qualification has been completed effectively.

Regarding documentation the meaning of the EC statement is not clear, in particular this one ''compliance with the latest requirements''.

4.6 References

[1] EGP 4101-6-940396 – March '95 ''Basic Concept of safety of Temelin operation''

[2] EGP 4101-6-990097 – 07/99 ''basic information for equipment qualification 2° step complex program''.

[3] EGP 4104-6-980018 – 12/98 ''Objectives, requirements and selection of equipment for qualification 1° step complex program''.

[4] EGP 4104-9-980020 – 12/98 ''Equipment safety classification step 1° complex program''

[5] IAEA Design safety series No. 50-C-D ''Codes on safety of NPPs design'' 1988

[6] IAEA Safety Guides 50-SG-D11 ''General design safety principles for NPPs'' 1988

[7] IAEA Safety Reports Series n. 3 ''Equipment qualification in operational NPPs'' 1998

[8] IAEA safety series No. 75 INSAG 3 ''Basic safety principles for NPPs'' 1998

[9] POSAR rev. 1 - 1999
Czech presentation to trilateral meeting in Prague, March 2001

[10] Working Paper Summarising The Outcome Of The Expert Mission With Trilateral Participation Established Under The Melk Protocol (Chapter IV), TREN/CW/sdp/2001-28, May 16, 2001

4.7 Attachment 1: Questions/Clarification on Issue 19: Environmental and Seismic Qualification of Equipment

(Following the presentation in Prague on March 15, 2001)

GENERAL

During the presentation of this issue in Vienna Austrian Experts clearly asked for information about the management of this issue in the frame of the licensing process and in particular with respect to the authorisation for commissioning, as this is an open issue.

The presentation in Prague does not give any relevant answer in this respect. Therefore the following important questions arose and should be discussed within the bilateral meetings devoted to this issue to complement the information received.

Q1) NPP role, management, coordination and responsibility:

- How does the Operator manage and co-ordinate the activity of the EQ Complex Programme? In which way does the Operator co-ordinate the work of the support organisations (REZ, S&A, ...) the contributions of eastern and western suppliers, the activities of test laboratories?

- Which QA programme has been established for the implementation of the EQ Complex Programme to ensure by appropriate procedures and by a system for independent review, audits, evaluation of activities which affect work quality, etc., that work is performed and documented consistently?

- How does the Operator manage the licensing process for EQ issue with respect to SUJB and what is the current status?

Q2) SUJB role and actions with respect to EQ issue:

- What has been the position of SUJB in allowing the startup of the NPP having an equipment qualification not fully established? How was the safe operation justified by the Operator in order to support the request of authorisation for fuel loading?

- Which conditions have been put forward by SUJB to the Operator in terms of maximum deadline and action plan, for resolution of this issue and justification for this interim phase?

- What specific surveillance has been put in place by SUJB in order to monitor the implementation of EQ Complex Programme by the Operator and the current status of licensing of that programme and related results?

SECTION 3 (of the presentation)

Q3) This question aims to get complementary information to understand the frame in which the current EQ Complex Programme is performed.

New design requirements arising from many facts like:

- the review of seismic design input for Temelin NPP took place in first half of nineties
- amendment to basic design for seismic qualification from EGP [1] in 1995 – POSAR [9] par. 3.2.1.1 /p. 3.2.1.6)
- important plant modifications
- new rules and requirements from SUJB not in force at the beginning of construction, etc.

have been introduced to the Temelin NPP after its first conception and design, as consequence a review of design acceptability of already installed equipment had to be performed.

To which extent was the acceptability of already installed equipment (mechanical, electrical, etc.) reviewed and which were the results in terms of acceptable equipment and not acceptable equipment and subsequent actions?

During the trilateral workshops there was evidence that the demonstration of functional qualification of safety and relief valves of steam lines is still under application to the Regulator (see issue 10).

This question deals with correct design capability and requirements of equipment and the necessary conditions before going into the issue of qualification of equipment, which means demonstration of capability of equipment to perform safety functions at the end of life, after a seismic event, during DBE, etc. (covered by the above mentioned EQ Complex Programme).

Q4) How can be considered archived (as said during the presentation) the seismic qualification of all equipment of Temelin NPP requiring it, if in harsh environment it is still on-going the review of performed EQ process for equipment important to safety and the seismic test is just a step in the overall equipment test sequence?
What action will be taken if the equipment proves not to be acceptable for environmental qualification after a seismic event?

Q5) How have ageing mechanism been considered with respect to seismic qualification in mild environment (not mentioned during the presentation)?

SECTION 5 (of the presentation)

Q6) What is the current situation of qualification of equipment belonging to safety related systems and what are the conditions and schedule for the process to reassess and establish full qualification requirements for them?

SECTION 6 (of the presentation)

To complement the information given during the presentation

Q7) In which way has the sequence of tests been organised, and justified, such that the stress conditions (due to ageing effects, accident conditions, etc) are applied in the most conservative direction to reproduce synergetic effects between different degradation mechanisms (i.e. temperature ageing, corrosion ageing, radiation, cycling, vibration, etc.)?

Q8) Which vibration induced by external man made hazards are considered in the test qualification sequence?

Q9) Regarding post DBE qualification: The need for equipment important to safety to remain operational requires definition and justification of a time-span after an accident, as well as correlation with emergency procedures. How has this task been accomplished and what are the results?

Q10) Survivability: What is the list of equipment required to survive during Beyond Design Basis Events or SA (i.e. H_2 explosion) and on which basis has its survivability been verified against the special environmental conditions arising from these events?

SECTION 7 (of the presentation)

Q11) From Tables 3 and 4 it appears that the EQ issue is resolved for power and I&C cables in harsh and mild environment, batteries, inverters, recombiners, while it is in progress for other equipment like sensors, valve actuators, cable penetrations, DG, pumps and motors etc. What is meant by ‘’ongoing’’ in those Tables in quantitative terms?

What are the findings up to now in terms of installed equipment needing requalification or replacement?
What has been reviewed and approved up to now by SUJB with respect to EQ Complex Programme execution?

How are qualification reports and records managed? Are they available onsite for all qualified equipment?

5 Seismic Design and Seismic Hazard Assessment – Issue 07

Table of Contents

5.1 Introduction

Seismic hazard estimates based on historical earthquake data and other traditional methods tend to underestimate true risk. New methods based on geological evidence have been developed and are being applied in many countries, some of which have started reassessment of the seismic hazard for their nuclear power plants. The IAEA has taken this development into account in its guidelines issued 1991 and recommends i.a. dating the youngest activity of seismic faults as well as inclusion of paleo-seismological methods.

5.2 Identified problems

Seismic Hazard Assessment for Temelin NPP is not based on state-of-the-art methods; there are indications that the currently assumed Safe Shut Down Earthquake (SSE) of 0.1 g (maximum horizontal peak ground acceleration) might underestimate the true risk.

Although the Temelin site is considered to be in a low seismicity area, this does not mean that strong earthquakes do not occur - they are just more rare. This fact has been demonstrated by several heavy earthquakes in areas previously assumed to be of low seismicity: Gujerat, India (2001), Roermond, NL (1992), Tangshan, China (1976). A recent EU research project has shown that it is more likely than previously thought that areas of present day low seismic activity are affected by large earthquakes (Paleosis, 2000).

Two major faults (Jachymov/Hluboka and Blanice) pass Temelin NPP site within 5 and 13 km distance, some smaller ones are even closer but are adressed as unimportant. The Czech side argues that these faults are inactive. This statement is unproven, as no state-of-the-art dating methodology has been applied yet.

Temelin NPP is situated north of the fault-bounded Budejovice Basin in the vicinity of several faults. Czech investigations prior to the siting of Temelin NPP proved an active subsidence in the Basin and assumed a young tectonic activity of deep seated faults crossing the Moldanubian rocks. Czech investigations after site selection assume a tectonic inactivity since 600,000 years without application of state-of-the-art methods (paleoseismology, dating of fault activity).

An IAEA Site Safety Review Mission in 1990 (Gürpinar et. al. 1990) recommended investigations that permit dating of recent tectonic activity (e.g. paleoseismology) and that are used internationally for seismic hazard assessment (e.g. Belgium, France, Germany, Italy, Netherlands, Portugal, Spain as well as Japan and USA). The IAEA recommendations were not carried out with state-of-the-art methods. The study by ENERGOPRUZKUM (Simunek, 1994, English version 1995) that was done in response to the IAEA recommendations, has not been peer reviewed officially and does not meet international standards in tectonic investigations.

To evaluate the seismic hazard a special zoning of the area around Temelin was introduced, however the arguments for neither the identification of seismogenic structures nor the assumed maximum earthquake potential are sufficiently convincing. The main objection is that maximum observed intensities are taken as maximum potential intensities. Such a hypothesis assumes that the maximum potential intensities for time periods of 10,000 or more years have all been experienced in the much shorter period of historical anecdotal and instrumental records. No reason for such an unusual outcome has been provided. Furthermore, maximum stress drop, possible length of the active part of a fault and displacement during an earthquake event were not examined.
Evaluation of the seismic design is not yet completed. In the course of other assessments, it became clear, however, that the Czech regulations for seismic qualification are a combination of parts of codes from different countries and the combination has not been assessed to determine whether the combination provides for conservative results.

5.3 Solutions to Identified Problems

This very complex and important issue merits a specific workshop, during which site relevant studies including the ENERGOPRUZKUM study, state-of-the-art methods such as paleoseismology and new insights on slow active faults gained from recent EU financed research projects should be discussed. This could also clarify the question of what are state-of-the-art methods to be applied in practice vs. scientific developments and interests. The Austrian and Czech sides came to an agreement to organise such a workshop on seismic issues during the last six months of the year 2001. The workshop is intended to concentrate on site seismicity and seismic design, and should involve geoscientists and structural engineers.

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

None of the meetings within the trialogue were devoted to the seismic issue. However, the Czech side made available a Report “NPP Temelin construction site – supplementary geological and seismological surveys (part A – Tectonics, part B – Seismic Risk)” by Energopruzkum, Praha, 1995. It is on the basis of this Report and the information given in the POSAR, that the Austrian position was developed.

5.4.1 Seismogenic Zones

The assessment of seismicity does not follow common EU-practice nor the recommendations of IAEA safety guide No. 50-SG-S1 which require the use of state-of-the-art methods. The methodology described in the reports by Energopruzkum (Simunek et al., 1995) is also not in accordance with the recommendations given by an IAEA site safety review mission (Gürpinar et al., 1990).

Some examples:

- Only one tectonic model of seismogenic zones was used in the original hazard assessment. This contradicts § 401 of the IAEA recommendation. In German practice different teams of experts perform hazard assessments, considering also a worst case scenario. During the ongoing Environmental Impact Assessment a model comparison was presented, which however has deficiencies in method and substance.

- The requirements in the chapter “Identification of seismogenic structures” (§ 409 to § 417 of the IAEA recommendations) are only partially fulfilled, because there is no clear or sufficient description of the seismogenic structures in the documents presented so far.

- Because the location and hypocentres of historical earthquakes are rather uncertain, greater tectonic areas – and stronger reference earthquakes – must be considered.

5.4.2 Earthquake Intensities

- The proposed method of fault activity assessment (determination of I_{0max} and M_{max} for single faults and zones) is unique, unclear and was never subjected to peer review on the international level.

- Observed intensity in most of the seismic regions of the model is equated with the maximum potential intensity, which is not correct even from a simple statistical point of view. Maximum credible intensities should be estimated at least one degree higher than observed intensities.

- The relation between intensity and ground motion calculated on the basis of following a global mean value (Murphy, 1977) is not conservative as compared to e.g. French SIN (1981) or Russian PNAE (1989) standards.

- The calculation of the attenuation curves of the seismic intensities (from the epicentre to the Temelin-site) are not transparent (neither in Simunek et al. 1995, nor in the POSAR). It is not explained how the curve (sometimes enveloping the data points, sometimes crossing them) was calculated, no confidence limits are given. Standard methods would have resulted in different attenuations and therefore different intensities at the site.

5.4.3 Fault Activity

Assessing the earthquake hazard in low seismicity regions of Europe and identifying the causative faults was considered to be difficult in the past because of the low number of strong earthquakes and the relatively short period of instrumental monitoring and historical information. A state-of-the-art evaluation of the maximum earthquake potential as well as an estimation of magnitude and recurrence time of these (pre-historic) events is now possible by identifying paleo-earthquakes along selected active faults.

Examples of deficiencies are:

- The paleoseismologic investigation (recommended by IAEA Site Safety Review Mission 1990) was not carried out, statements on tectonic inactivity are not backed up by relevant investigations and determination of the age of the youngest movements.

- The few, short trenches carried out have not been placed across the relevant tectonic structures (geomorphological scarps, fault-traces), none of them across the faults nearest to the site.

- The Czech ENERGOPRUZKUM report on the tectonic situation neglects the results of earlier investigations on recent crustal movements by Czech geoscientists (Kutina, 1976; Vyskocil, 1975; Vyskocil & Kopecky, 1974; Vyskocil & Zeman, 1979).

- Although the ENERGOPRUZKUM report (1995) lists some observations that describe features normally attributed to active tectonics (e.g. faults and thrusts), the report draws conclusions that differ from these observations.

- The claim of “tectonic inactivity since 0.6 million years” (this number differs within the report) is not based on results of adequate methods, e.g. no geochronological dating of movements.

5.4.4 PSHA

The probabilistic seismic hazard assessment (PSHA) is not state-of-the-art. The purpose of PSHA is to reflect the range of technically credible opinion within the knowledgeable scientific community. Not withstanding this fundamental purpose, the PSHA performed for Temelín includes no uncertainty in seismic zonation — only one zonation map is used, and that map is more than 20 years old now. Within the range of credible models of attenuation of seismic energy with distance, three models were used whereas state-of-the-art studies use five or more such models.

The uncertainty in the maximum earthquake intensity (I_{max}) was limited to 0.5-1.0 MSK-64 units, whereas state-of-the-art studies reflect a greater range of values than this. In addition, there are errors in transcribing I_{max} data from the text discussion to the table, which was used in model quantification.

In short, rather than by incorporating the full range of scientific opinion, the Temelin PSHA has constrained the model artificially to a narrow uncertainty band in several key elements of the model. In one fundamental area of the model the PSHA reflects no uncertainty at all.

5.4.5 Seismic Design

Not much time could be devoted to the evaluation of seismic design and seismic equipment qualification as yet. Nevertheless, one aspect the Austrian experts came across should be mentioned here:

In the presentation of the Czech experts 8 normative Codes were named for the seismic qualification and 5 normative regulations for the component qualification. A question concerning the normative basis of the seismic component qualification started an investigative effort on the Czech side and was later answered in a written format. According to this the demonstration of the seismic strength is performed with the following methodology:

The complete component and piping system is calculated as piping statics using the calculation code SYSPIPE Cod, version 231c from FRAMATOM_FRAMSOFT/CSI, certified for the Czech Republic by SUJB. Based on these calculations, the stresses within the components are calculated according to the Russian Code G [1]. For a T-type pipe this can be performed using two different methods:

- a simplified calculation procedure: attachment 5 in [1]: calculation of typified components and devices (this attachment is of recommendatory character), paragraph 2.3: piping systems with low temperature loads

- in depth calculation method: attachment 5 in [1]: calculation of typified components and devices (this attachment is recommendatory character), paragraph 2.9: calculation of stresses in a T-type branch using more precise methodology

In a third step FEM calculations of the component can be used for an even more precise calculation of the stress distribution. The resulting stress categories are compared with the respective limiting values in the ASME-code, Section III, NB.

This means that a combination of three different code regulations with completely different origin and development history is applied. This requires a critical analysis, e.g. a comparison of the initial assumptions (selection of the design earthquake, validation of the response spectra, load assumptions) within the different codes. This is necessary because the individual national normative codes always assume a relation between the different steps. For instance, in case of very precise or very conservative initial parameters the required safety margins can be rather low, while these would lead to completely wrong conclusions in case of simplified or non-conservative initial parameters. Especially the selection of the maximum credible earthquake according to different assumptions in the codes and the requirement of precision in the calculation of response spectra strongly influences the required safety factors for the definition of the allowable stresses.

A comparative analysis especially with respect to conservatism of the methodology used in the commissioning of Temelin NPP has not been performed up to now.

5.5 Technical Arguments

5.5.1 Regulations and standards for seismic hazard evaluation

The Czech side claims that “all relevant Czech regulations and internationally accepted codes and standards, etc.” were respected (Czech Republic Report to the Expert Mission etc., 2001, p. 27). This is not the case: The former Czech regulation – valid at the time of siting of Temelin NPP - excluded regions of subsidence for the siting of nuclear power plants. The Budejovice Basin is an area of subsidence according to Czech scientists (Vyskocil & Kopecky, 1974; Vyskocil, 1975).

In applying the deterministic approach for assessment of a maximum credible earthquake (MCE) Czech studies assumed an epicentral intensity of I_0=8°MSK for the maximum observed historical earthquake (Neulengbach 1590) whereas the scientifically accepted intensity was I_0=9°MSK (Gutdeutsch et al.). Therefore the local intensity in Southern Bohemia can be derived to be in the range of I=6.0°-6.5°MSK (according to historical observations). Adding the usual one degree for the Maximum Credible Earthquake results in I_{max}=7.5°MSK as a minimum level. Some authors even add a value of 1.5°MSK. Thus the Czech approach adding just 0.5° is unusual and cannot be considered to be sufficiently conservative.

Simunek et al. assume a correspondence of an intensity of I=7°MSK to a PGA (maximum horizontal peak ground acceleration) of 0.1 g (Simunek et al, 1995). This is not conservative since it reflects only a mean PGA value for the whole world (Murphy, 1977). Some standards (e.g. French but even Russian – in contrast to “Russian practice”) - are even more conservative, thus correlating intensity levels with higher g-values of PGA.

The Neulengbach event of 1590 was not mentioned in the catalogue presented to IAEA Site Safety Review Mission 1990 - the catalogue started with the year 1593! Thus a critical discussion with the review mission was avoided.

5.5.2 Fault Activity

The term “diffuse seismicity” is incorrect for the region. Latest results by Austrian seismologists (Lenhart 2000) on microseismicity in the Austrian part of the Bohemian Massiv confirm earlier results by Czech scientists: There is a good correlation between microearthquakes and faults.

Evaluation of capability of faults was performed with a method used only by Czech investigators for Czechoslovakian nuclear power stations. IAEA 1990 recommended state-of-the-art methods (e.g. geomorphology and paleoseismology), supported by determination of the age of youngest movements. The reports by Simunek et al., 1994/1995 (Energopruzkum) do not show the necessary data (Simunek et al., 1995). In situ geological and geomorphological observations and geophysical investigations were proposed by IAEA (several years after siting!) but no follow up mission has taken place yet (Gürpinar, pers. comm.).

The tectonic activity of Jachymov Fault zone (Hluboka fault scarp is only a section) was not investigated with state-of-the-art methods (paleoseismology). Drilling and geoelectric measurements are not sufficient. IAEA explicitly recommends appropriate techniques to ascertain when movement last occurred.

5.5.3 IAEA 50-SG-S1 recommendations compared to Energopruzkum's study

5.5.3.1 Near region investigations

IAEA

328. Investigations … should typically include (2): Neotectonic studies to determine the latest movements of faults. To reach this goal, geomorphology, boring, trenching pedology, studies of the Quaternary deposits and age dating may be necessary, as well as paleoseismicity, geophysics, geochemistry, geodesy and in situ stress measurements.

Energopruzkum provided some evidence of quaternary river terraces in the style geographers have done this for decades – following Penck & Brückner's scheme (Günz, Mindel, Riss, Würm), which was established almost a century ago. Meanwhile however, different techniques are available to effectively date the sediments. Boreholes and geoelectric lines were carried out, but not the trenches across the fault traces.

5.5.3.2 Site vicinity investigations

IAEA

329. Investigations of the site vicinity shall be conducted to define in greater detail the neotectonic history of the faults and to identify sources of potential instability.
330. The investigations should provide the following: (2) A neotectonic history showing the age and amount of fault displacement based on trenching and age dating, as appropriate.

Energopruzkum presented some schematic geological and geomorphological cross sections. State-of-the-art trenches, as published in international journals and demonstrated on the web-pages of several institutions, were not performed. The information gained from some few artificial outcrops cannot be compared with information to be expected from paleoseismological trenches such as those already performed across faults of the Lower Rhine Embayement (around 75 m long and reaching a depth of 5 m). Besides, the artificial outcrops presented in Simunek et al. were not located directly across the fault scarps (Simunek et al., 1995). No age datings were carried out.

5.5.3.3 Construction of a regional seismotectonic model

IAEA

401. ... In its construction, all existing interpretations of the seismotectonics of the region that may be found in the available literature should be taken into account.
403. … Seismogenic structures may exist without recognized surface or subsurface manifestations and because of the time-scales involved, e.g. fault displacements may have long recurrence intervals with respect to seismological observation periods.
406. … any tendency to interpret data only in a manner which supports some preconception should be avoided.

407. … when it is possible to construct alternative models which explain the observed seismological, geophysical and geological data equally well ... the final hazard assessment should take into consideration all such models.
408. The most important of the parameters which have to be evaluated is the maximum earthquake potential that can be associated with either a seismogenic structure or a zone with diffuse seismicity.

Energopruzkum used one seismotectonic model, different from that in the POSAR.

Alternative models (worst case, preferably done by different teams) were not taken into account. Faults with known tectonic activity outside the Czech Republic end on the presented map before reaching Temelin. In contrast, earlier studies (Grünthal et al., 1985) show young activity of shear zones even within the Bohemian Massif.

Seismicity is now called diffuse – in contrast to earlier findings by Czech authors, stating that faults and microearthquakes are in agreement (Kutina, 1976). Their studies were published well before siting decision and can therefore be considered not to be influenced by economic interests. This result corresponds to recent findings by the Austrian Seismological Survey: A map of 1741 registered earthquakes below the threshold of human perception between 1995 and 2000 demonstrates a correlation between faults and microearthquakes (Lenhardt 2000).

5.5.3.4 Levels of design basis ground motion

IAEA
503. The SL-2 level corresponds directly to ultimate safety requirements. This level of extreme ground motion represents the maximum level of ground motion to be used for design purposes.
520. After the design base intensity at the site has been established, the corresponding maximum ground acceleration or velocity should be obtained using relevant empirical relationships. In the selection or generation of an appropriate relationship it is recommended that the dispersion of data, which can be very large, be taken into consideration.

Energopruzkum
When determining the ground acceleration based on the intensity of earthquakes Energopruzkum follows only the mean value for the world (Murphy, 1977). Although 0.1 g is the minimum level recommended by IAEA for any site the Czech Republic Report to the Expert Mission etc. (2001) insists on a hazard value less than 0.1 g (6.5°MSK).

5.5.3.5 Potential for surface faulting at the site

IAEA
603 (1) ... In highly active areas, where both earthquake and geological data consistently reveal short earthquake recurrence intervals, periods of the order of tens of thousands of years may be appropriate for the assessment of capable faults. In less active areas, it is likely that much longer periods may be required.
606. When faulting is known or suspected, investigations should be made which include stratigraphical and topographical analyses, geodetic and geophysical surveys, trenching, boreholes, age dating of sediments or fault rocks, local seismological

investigations and any other appropriate techniques to ascertain when movement last occurred. Linear topographical features …should be investigated in sufficient detail to explain their cause.
607. The use of more than one technique for estimating the age of movement of faults will improve the reliability of the assessment.

Energopruzkum used its own method of fault assessment, which was never published and peer reviewed in an international journal. Faults were not properly investigated. The information obtained from boreholes placed near a fault is insufficient for the determination of fault activity. Geoelectric profiles were carried out but they were not analysed to select suitable places for trenching across geomorphological scarps or faults. No age dating was performed to ascertain when movement last occurred. Earlier geodetic measurements suggesting an active subsidence of Budejovice Basin were not taken into account.

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6 Cluster "+28.8 m Level"

Issue 8: Main Steam Line and Feedwater Line Breaks
Issue 10: Qualification of main steam line safety and relief valves for two-phase and water flow

Table of Contents

6.1 Introduction

The intermediate building is located between the containment building and the turbine hall. At the level of +28.8 m the high energy piping of the main steam is installed and therefore the area outside the containment must be considered sensitive to failure consequences. Special attention must be paid to the design in order to prevent damage to adjacent lines and other safety-related components resulting from high energy line breaks (HELB).

Adequacy of qualification of relief and safety valves is also questionable. These valves are connected to the main steam lines and installed in the intermediate building as well (see Issue 10 below).

In case of inadequate design provisions damage or malfunction of the piping and components could cause severe accidents with large releases of radioactive material.

6.2 Issue 8: Main Steam Line and Feedwater Line Breaks

6.2.1 Identified Problem

The main steam lines and feedwater lines are located within compartments A820 and A826 1/2 at the +28.8 m level in parallel between the main isolation valves and the penetration leading into the containment over a distance of almost 30 meters. There is no physical separation of these pipes from each other (by e.g. concrete walls) and from other equipment relevant to safety. In case of a rupture of one single steam line or a feedwater line consequential damage of the adjacent steam and feedwater lines as well as of other safety-relevant equipment due to pipe whip and/or jet impingement effects by discharged effluents is most likely the case. This could trigger an accident sequence e.g. with reactivity excursion resulting in core melt, direct containment heating and in large radioactive releases

IAEA issue AA5 provides some information on recriticality events in case of steam line breaks, depending on the assumptions made for the accident analysis treating comparable situations. The POSAR only provides information about the consequences of rupture of a single steam line.

The Czech approach to resolve high energy pipe break/whip and possible consequential damage to adjacent high energy pipes, components and equipment relies on calculations, based on a rarrow range of possible events.. The results of these calculations suggest that there would be no need for additional protective measures. Therefore only two pipe whip restraints per pipe are installed (one at confinement penetration and the other at the partition wall to the machine hall; total pipe length without pipe whip restraint about 30 m). Although necessary, these calculations do not provide sufficient evidence that the problem has been resolved, since they do not take into account “unexpected events” (Czech Rep. Report, 2001), i.e. a broader range of assumptions including e.g. transient and dynamic effects.

Although specific inspection, operation of pipes under appropriate chemical regime and other measures mentioned by the Czech side are positive features, they cannot prevent the possible consequences of e.g. water hammers.

In order to cope with such events and resulting accident conditions in NPPs in the EU, in addition to the introduction of break exclusion concepts, pipe whip restraints along these lines and physical pipe separation installations are applied. Since no physical separation and only two pipe whip restraints (with questionable robustness) per line are installed on the +28.8 m level, multiple pipe failure and consequential failures (e.g. of valves) cannot be excluded. Such damage could result in a large release of radioactivity, since there are most likely no compensatory measures possible.

The effectiveness of the installed pipe whip restraints was not demonstrated and concern remains about the possibility of multiple steam line breaks (and/or multiple feedwater line breaks) and consequential failures. Unless a comprehensive analysis of multiple steam line (and feedwater line) rupture and of consequential failures provide evidence regarding control of such accidents, a subsequent severe accident situation with serious consequences must be anticipated.

According to the latest Czech information future efforts include an extended assessment of all relevant damage mechanisms, qualified non destructive examination by mechanised ultrasonic testing and application of a well founded concept of the “Break Exclusion". However, these efforts are generally understood as preconditions for start-up (compare e.g. Issue 22, NDT) and may reduce the probability of failure of major piping systems, but they cannot replace the necessary hardware measures generally applied in comparable situations.

In summarising, there is evidence that in the assessment of the situation on the +28.8 m level the Czech side did not demonstrate the complete safety case through adequate and exhaustive analyses and has not demonstrated that multiple steam line breaks with serious consequences for plant safety can be managed or excluded. Necessary hardware measures have not been taken.

There was also no evidence on how the issue has been and is managed during the licensing process. The role of the Licensing Authority - in particular the specific bases for the final design approval (e.g., the criteria applied, the analyses required, etc.) could not be identified.

6.2.2 Solution to the Identified Problem

Austrian experts consider the following steps mandatory to resolve the problem, however the list may not be exhaustive:

Comprehensive analyses must be performed of multiple steam line and feedwater line ruptures providing detailed information on

- reactivity control during such accidents
- the subcooling effects on the pressure vessel and the involved steam generators and related danger of pressurised thermal shock

The essential steps are:

A demonstration of the complete safety case providing information about:

- the pending results of assessments of water-hammer and dynamic effects in feedwater and steam lines
- results of the above mentioned comprehensive analyses of multiple steam line and feedwater line ruptures. These results should include information about acceptable limiting break conditions and related direct consequences, which can be managed as DBA.
- any envisaged reconstruction or re-positioning/re-routing of safety-important components/equipment and of the steam lines/feedwater lines at the +28.8 m level
- demonstration of the robustness and adequacy of installed pipe whip restraints
- information about erosion-corrosion prevention and mitigation programmes and their implementation

A reconstruction of the +28.8 m level, based on complete and adequate re-assessment is required to physically exclude multiple steam line breaks and to exclude detrimental consequences for a broader range of possible events.

Re-routing of emergency feedwater lines is required if they are endangered by main steam line/feedwater line pipe whip (NRI, 2000).

Beyond that, it is essential that the operator implements the inspection programme that has been developed and that the safety authority ensures that the programme is implemented. The requalification for service is an essential part of the evaluation taking into account the complete equipment history and the ISI inspection evaluation (see also NDT, issue 22).

Timeline: Neither for Temelin Unit 1 nor Unit 2 would European state-of-the-art practice permit operation or even fuel loading before resolution of the HELB issue on the +28.8 m level.
All the necessary analyses could be completed within one year; the extent and the time needed for the adaptive measures resulting from the analyses cannot be estimated without knowledge of the results of the relevant analyses.

6.3 Issue 10: Qualification of main steam line safety and relief valves for two-phase and water flow

6.3.1 Identified Problem

The installed relief valves (BRU-A) and main steam line safety valves (MSSV) are inadequately qualified. Thus it cannot be excludet that the valves remain stuck open in case of steam and water mixtures flow and thereby initiate an accident. This could trigger an event sequence resulting in a severe accident with large releases of radioactive material.

In addition, isolation valves on the main steam lines upstream BRU-A, which could mitigate the adverse consequences of a stuck open BRU-A, have not been installed at the Temelin NPP.

The concern originates from the fact that for the Temelin MSSVs and BRU-As no plant or equipment specific functional qualification tests were performed.

The Czech side advocates a similarity approach for Temelin: Qualification results from different valves (parent design and smaller size) were extended to (candidate) valves corresponding to those installed in Temelin. For this analysis qualified parent type valves for MSSV and BRU-A were used; these parent valves are from the same manufacturer but differ in model and dimension (the latter was qualified for Mochovce NPP).

It has not yet been possible to verify that all conditions required for the application of the similarity approach are fulfilled, in particular with regard to the specific operating conditions of the valves installed at the Temelin NPP.

Without proof and a complete qualification documentation on the Temelin BRU-A and the MSSV the operator cannot confirm that the Temelin BRU-A and the MSSV are qualified for two-phase and water flow regimes.

In particular, in accordance with ASME[1], QME-1, 1994, (QV-8000) the following qualification documentation is required:

a) qualification plan for parent valves (QV-8200),
b) functional qualification report for parent valves (QV-8310),

[1] ASME code, QME-1, Qualification of active mechanical equipment used in NPP, section QV (qualification of valves), 1994.

c) application report of the Temelin candidate valves (QV-8320).

These documents were not made available at the meeting in Rez.

An Austrian request for the documentation on the qualification of BRU-A has led SUJB to conduct an audit in Mochovce, and to present a report on this audit at the meeting in Brussels. There however it turned out that - contrary to the information provided earlier by the Nuclear Research Institute (NRI) at a workshop on issue 10 (Workshop Rez, 2001) and the representative of the Regulatory Body - documentation about the qualification of the valves is not yet available. An audit of the BRU-A valve similarity proof with the main designer was then still pending. Thus in contradiction to the information provided at the meeting in Rez it has to be concluded that BRU-A valves for Temelin are not yet qualified.

This is supported by NRI information obtained in Rez and Brussels as well as from independent sources that the main designer Chekhov is still performing functional qualification tests for BRU-A valves of the Temelin size. It is essential that Chekhov BRU-A qualification test results be made available to Temelin for proof of the adequacy of the qualification approach intended for Temelin.

The situation for the MSSV qualification is apparently similar, these valves have also not yet been qualified. However Chekhov is changing the same type of valves at the WWER 1000 reactors in Kozloduy for qualification reasons (NEI, 2001). Proof of MSSV qualification for Temelin is essential in the light of MSSV replacements in Kozloduy.

6.3.2 Solution to the Identified Problems

Full demonstration is requested of the functional qualification approach taken for the main steamline safety and relief valves of NPP Temelin by providing qualification documents according ASME QME-1 1994.

The results of ongoing plant-specific BRU-A qualification tests performed by the designer Chekhov (information from NRI representative (Workshop Rez, 2001)) should be monitored and taken into account in the choice of measures to resolve this problem.

Installation of qualified isolation valves on main steam lines upstream of BRU-A is recommended (European practice, e. g. United Kingdom Sizewell B), but it is necessary in case of insufficient qualification of BRU-A.

Timeframe:
For both theTemelin Unit 1 and Unit 2 European state-of-the-art practice would not permit operation or even fuel loading before resolution of the valves qualification issue.
Within one year all the necessary analyses and measures could be completed.

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

6.4.1 Issue 8: Main Steam Line and Feedwater Line Breaks

Category of Deviation: high safety significance/large or moderate deviation

IAEA Ranking of this Safety Issue (IAEA, 1996a): Category II

6.4.1.1 Germany

In principle, the proof of safety of the pipes of the secondary circuit in German NPPs relies on break exclusion. In order to be on the safe side however, physical separation is also extensively implemented, particularly in newer plants[2].

The emphasis on physical separation is clearly shown in German nuclear regulations. It is, for example, required that valves relevant for containment isolation as well as pipe outlets from the containment are protected against the consequences of pipe breaks (jet and reaction forces, missiles)[3]. Isolation valves have to be physically separated[4].

In a typical, modern German PWR of the 1.300 MWe class, the feedwater and steam lines belonging to the four steam generators are strictly separated inside the reactor building. They all leave the containment on one side, towards the turbine hall. The steam valves (safety, relief and isolation valves), together with the regulating station for feedwater, are installed outside the reactor building for each line.

The pipes are installed side by side, but enclosed in four separated concrete ducts. Steam and feedwater lines are also separated from each other. Thus, there is no possibility for interaction; in particular, a break of one steam line cannot damage valves belonging to another steam line[5].

Only in older German reactors physical separation is less strict. In any case improvements have been introduced through backfitting, also in the Netherlands (Borselle NPP with Siemens PWR).

The Stade PWR (the second oldest German PWR) has been backfitted with steam isolation valves. They are mounted into the four steam lines inside the containment, well separated from each other and from other steam valves.

Generally, the emphasis is strongest on physical separation in Germany. Whereas it is considered important in other Western countries, there is more reliance on pipe whip restraints in some of their plants [6].

6.4.1.2 France

French PWRs are also designed using extensive physical separation regarding their steam and feedwater systems. French regulations require precautions against the effects of pipe breaks, particularly regarding isolation valves [7].

The design of older French PWRs (model 900 MWe) regarding steam and feedwater lines is very similar to that of German PWRs with 1.300 MWe. The pipes belonging to the three steam generators leave the reactor building side by side, but are separated by concrete walls. Steam safety, relief and isolation valves for each line are in different

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[2] In-Depth GRS Analysis of Seven Selected Safety Issues Relating to the Temelin NPP, Draft; Gesellschaft für Anlagen- und Reaktorsicherheit mbH im Auftrag des BMU, August 2000
[3] Sicherheitskriterien für Kernkraftwerke; Bundesministerium des Inneren, 21. Oktober 1977, #8.4
[4] RSK-Leitlinien für Druckwasserreaktoren; Reaktor-Sicherheitskommission, 3. Ausgabe vom 14.10.1981, zuletzt geändert 1996, #5.6
[5] Kernkraftwerke; Technischer Verlag Resch/Verlag TÜV Rheinland, Köln 1986
[6] See Footnote 1
[7] Lettre d'Orientation SIN No 576/78, Ministre de l'Industrie, 3 Mars, 1978

compartments. Steam lines are located above the feedwater lines and also separated by concrete walls. The layout is shown in figure I[8].

In French NPPs of the 1.300 MWe model, a different solution has been adopted: Steam and feedwater lines belonging to the four steam generators leave the reactor building corresponding to the positioning of steam generators inside. There are two pairs of outlets on opposite sides of the building; the distance between neighbouring outlets is about 25 m. The valves are located outside the reactor building near the penetrations (see figure II)[9]. Between reactor building and turbine building, steam and feedwater lines are protected by a light shelter.

6.4.1.3 United Kingdom

For the British PWR Sizewell B (1.188 MWe, four loops), the layout of steam and feedwater lines as well as steam valves is comparable to that of modern German PWRs or French PWRs of the 900 MWe model.

Steam lines and feedwater lines leave the containment on one side. Steam lines are above feedwater lines; the lines are grouped in two pairs. They enter the steam and feedwater cells in parallel after leaving the containment. The cells contain safety and relief valves as well as isolation valves[10].

There is no complete physical separation, but there are two separated cells, each containing one pair of steam or/and feedwater lines[11]. By the pipe whip restraints installed on the lines, damage of the other line can be excluded.

6.4.1.4 Spain

Modern Spanish PWRs like Vandellos-2 (1.052 MWe, three loops, commercial start 1988) make use to a considerable extent of physical separation of steam and feedwater lines, with again a layout similar to modern German PWRs or French PWRs of the 900 MWe model.

The steam and feedwater lines at Vandellos-2 leave the containment on one side and enter the so-called penetration building. Steam lines are located above feedwater lines. Safety, relief and isolation valves are located in this building. The distance from the containment wall to the steam isolation valves is about 10 m. All lines are in individual chambers, separated by concrete walls (ceilings)[12]. Thus, there is no possibility of failure propagation in case of, for example, steam line break between containment and isolation valve.

[8] 900 MW Nuclear Power Plants; EdF, 3rd edition, May 1983
[9] Centrale de Flamanville, Tranches 1 et 2; EdF, Juillet 1983; and: 1300 MW Nuclear Power Plants; EdF, 3rd edition, June 1983
[10] Layfield, S.F.: Sizewell B public inquiry; Department of Energy, London, 1983, 30
[11] Layfield, S.F. (see footnote 10), C37.5; and: The British PWR; Special Publication, nuclear engineering international, undated (ca. 1988)
[12] Vallejo, J.: Spain's nuclear industry achieves maturity with its third generation of PWR; nuclear engineering international, September 1986, p. 32

6.4.1.5 Summary

Pipes without physical separation or whip restraints, mounted in parallel over a distance of 20 m and more do most certainly not conform to European practice for newer plants. Most plants have full or at least partial physical separation and much shorter distances between whip restraints.

6.4.2 Issue 10: Qualification of main steam line safety and relief valves

Category of Deviation: high safety significance/large or moderate deviation
IAEA Ranking of this Safety Issue (IAEA, 1996a): category III

WENRA Report stated for Temelin: Plant-specific safety demonstration for the functioning of the main steam relief valves and the main steam safety valves under dynamic loading with a steam-water mixture still has to be fully verified. This action is on the way. This function is needed to control specific primary to secondary leaks (WENRA, 2000).

Germany: Bleed and feed capability, and hence qualification for two-phase flow, is of particular importance for main steam line relief valves in German PWRs since secondary-side bleed and feed is the first preference as an accident management measure (Smith, 1999). The Positional Report of the German Nuclear Safety Standards Commission (KTA, 1996) states that all German operating PWRs, except Biblis A and B, have fully implemented secondary feed accident management option. Biblis A and B have applied for the license.

The relief valves of Biblis B have a motor operated shut off valve, which can be closed manually by the operator from the control room (Boneham, 1999).

Belgium: Tihange 2 is provided with a relief valve per SG which can be isolated by its associated isolation valve (Boneham, 1999).

France: N-4 (newest French series: 2 units at Civeaux and 2 at Chooz) - to limit the risk of water discharge through safety valves, a redundancy of the atmospheric dump system is implemented; each steam generator is equipped with two redundant safety-grade lines, each having a discharge valve and an isolation valve qualified for water discharge (Berger, 1996).

Great Britain: On Sizewell B the normal secondary relief valve is provided by a single PORV on each SG. This valve can be isolated by the operator if it is found to be stuck open or leaking, through the closure of an isolation (block) valve (Boneham, 1999).

Sweden: Ringhals 3 has one relief valve and 6 safety relief valves per SG. The relief valve has an associated isolation valve (Boneham, 1999).

Advanced reactors:

European Pressurised Water Reactor (EPR): For the EPR it is planned to install an isolation valve on the steam line upstream of the steam relief valve.

AP 600 - advanced passive PWR - (Boneham, 1999) has 3 safety valves and 1 PORV. The PORV can be isolated using its associated isolation valve.

Deviation from ASME code: The ASME code, QME-1, 1994, requires documentation of the qualification test results and the design: a) qualification plan for parent valves, b) functional qualification report for parent valves, c) application report of the candidate valve.

General requirements on equipment qualification and documentation are reviewed in the position paper on issue 19: Environmental Qualification of Equipment.

6.5 Technical Arguments

6.5.1 Issue 8: Main Steam Line and Feedwater Line Breaks

GRS:
Die für KKW-T entwickelte Rohrausschlagsicherung ist aus Sicht der GRS in ihrer Wirksamkeit eng bezogen auf idealisierte Bruchannahmen. Inwieweit die getroffenen Maßnahmen zur Beherrschung von Folgeschäden ausreichen, konnte von der GRS anhand der ihr vorliegenden Unterlagen nicht abgeschätzt werden. Jedoch hält die GRS das unterstellte Spektrum von Bruchannahmen und dadurch bedingter Folgeschäden für zu eng und daher die Konstruktion nicht für ausreichend robust. Darüber hinaus ist das Aufschweißen von Halteplatten auf die drucktragende Wand - obwohl nach den Regelwerken verschiedener Länder zulässig - aus Sicht der GRS keine zeitgemäße technische Ausführung für die Lösung derartiger Fragestellungen." (GRS, 2000)

Wenra:
Some specific issues, however, e.g. protection against postulated steam piping or feedwater line rupture on the +28.8 m level, need further consideration" (WENRA, 2000).

USNRC:
USNRC regulations dealing with high energy pipes design are applied in Spanish PWRs made by Westinghouse like Vandellos-2 (1.052 MWe, three loops, commercial start 1988). Nevertheless, there is a high degree of physical separation of steam and feedwater lines, with a layout similar to modern German PWRs or French PWRs of the 900 MWe model. (see details above).

Only based on a fully demonstrated comprehensive safety case including limiting cases, adequate preventing and mitigating measures can be considered, selected and implemented.

Separation of main steam and feedwater lines and installation of pipe whip restraints are the practice dominant within the EU.

Reconstruction of the +28.8 m level should be seriously considered for a plant starting operation 2001. From a technical point of view the preventive and mitigating measures taken so far appear to be insufficient to solve the problems indicated.

Reconstruction at NPP Borselle (CNS, 1998):

The following Reconstruction of main steam and feedwater lines was performed at NPP Borselle (Netherlands) with a Siemens PWR:

In the late 1990s replacement of the existing main steam and feedwater lines inside the containment and annular space (between the inner and outer containment) and partially in the turbine hall by

- *qualified "leak before break" piping ;*
- *steam flow limiter at the containment penetration location and*
- *guard pipes around steam and feedwater lines in the auxiliary building*

<u>6.5.2 Issue 10: Qualification of main steam line safety and relief valves</u>

The lack of qualification of main steam line safety (MSSV) and relief valves (BRU-A) for (high velocity) two phase water flow and water flow is a generic safety issue for all WWER-1000/320 reactors, as well as a specific issue applicable to Temelín.

In the WWER-1000/320 design, there is one BRU-A and two MSSV on each of the four main steam lines, installed upstream of the main steam isolation valve (MSIV) in the area of the intermediate building outside of the containment. The BRU-A can not be isolated against the atmosphere should it open and subsequently fail to close (see Appendix 2: Arrangement of MSSV and BRU-A valves on steam lines).

The German GRS, in 1993, concluded for Stendal NPP that as a result of accident analyses performed for Stendal, the BRU-As have to be designed for two-phase flow. Additionally GRS recommended to install an isolation valve upstream of BRU-A. The IAEA WWER-1000 safety issue book and the IAEA Temelin safety issues resolution report, both from 1996, also recommended to qualify the BRU-A and the MSSV for water and two phase flow and to investigate the possibilities to install a qualified isolation valve upstream BRU-A.

According to information presented at the Workshop in Rez, 26/27. Feb. 2001, tests for the Mochovce BRU-A were performed at the EdF-CUMULUS test facility and a prototype WWER-1000 MSSV was tested by Siemens KWU and Chekhov, the main designer of the valves. Referring to ASME code, QME-1, 1994, on design similarities and testing, Rez concluded that the Temelin BRU-A are qualified for two-phase and water flow. The similarity of Temelin MSSV with the one tested by Siemens/Chekhov cooperation were reported to be assessed by experts and therefore the Temelin MSSV is argued to be qualified for two-phase and water flow.

According to NRI all the test data as well as testing philosophy for the BRU-A is confidential property of the consortium which performed the tests. Temelin NPP as well as NRI could only quote documentation of the tests (Workshop Rez, 2001). The Austrian experts would have to contact the firms directly in order to receive more detailed information. There is some question as to how the requirement that a complete set of equipment qualification documentation has to be available at the plant can be fulfilled under these circumstances (for the general requirements on EQ documentation see issue 19).

Without providing a complete qualification documentation on the Temelin BRU-A and the MSSV in accordance with ASME, QME-1, 1994, which was requested (Workshop Rez, 2001) no proof was obtained that the Temelin BRU-A and the MSSV are qualified for two-phase and water flow.

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IAEA, 1996a — International Atomic Energy Agency (IAEA*), Safety Issues and*

	Their Ranking for WWER-1000 Model 320 Nuclear Power Plants, Extrabudgetary Programme on the Safety of WWER and RBMK Nuclear Power Plants, IAEA-EBP-WWER-05, March 1996.
IAEA, 1996b	International Atomic Energy Agency (IAEA), *Comparison of PSA Results With the Programmes of Safety Upgrading of WWER NPPs*, WWER-SC-152, Draft, Rev. 2, 15 May 1996.
IAEA, 1996c	International Atomic Energy Agency (IAEA), *Report of the Review of WWER-1000 Safety Issues Resolution at Temelin Nuclear Power Plant*, Temelin, Czech Republic, 11 to 15 March 1996, TC Project RER/9/035, IAEA-TA-2490, WWER-SC-171.
IAEA, 1996d	International Atomic Energy Agency (IAEA), *Report of an Expert's Mission to Review the Modernisation Programme of Rovno NPP Unit 4*, Rovno, Ukraine, 2 to 12 October 1995, TC Project RER/9/035, IAEA-TA-2486, WWER-SC-151, March 1996.
IAEA, 1997	International Atomic Energy Agency (IAEA), *Report of an Expert's Mission to Review the Modernisation Programme of Khmelnitzky NPP Unit 2*, Khmelnitzky, Ukraine, 10 to 14 June 1996, TC Project RER/9/035, IAEA-RU-6131, WWER-SC-178, February 1997.
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NRI, 2000	Nuclear research Institute Rez, *Response to the IAEA Document Safety Issues and Their Ranking WWER-1000 Model 320 NPPs for Temelin NPP*, No. 11315 T/R 3a, March 2000.
NUSIM, 1997	C. Lathuile, J. L. Serre, *Analysis and Qualification of the Steam Generator Relief Valves (BRU-A),* NUSIM´97 workshop, Nov. 19-21, Levice, Slovakia.
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6.7 Attachment 1: Background Documentation

6.7.1 Issue 8: Main Steam Line and Feedwater Line Breaks

Figure 1: Layout of Main Steam Lines – French 900 MWe Model

The layout for modern German PWRs is very similar – apart from the fact that they have four steam generators.

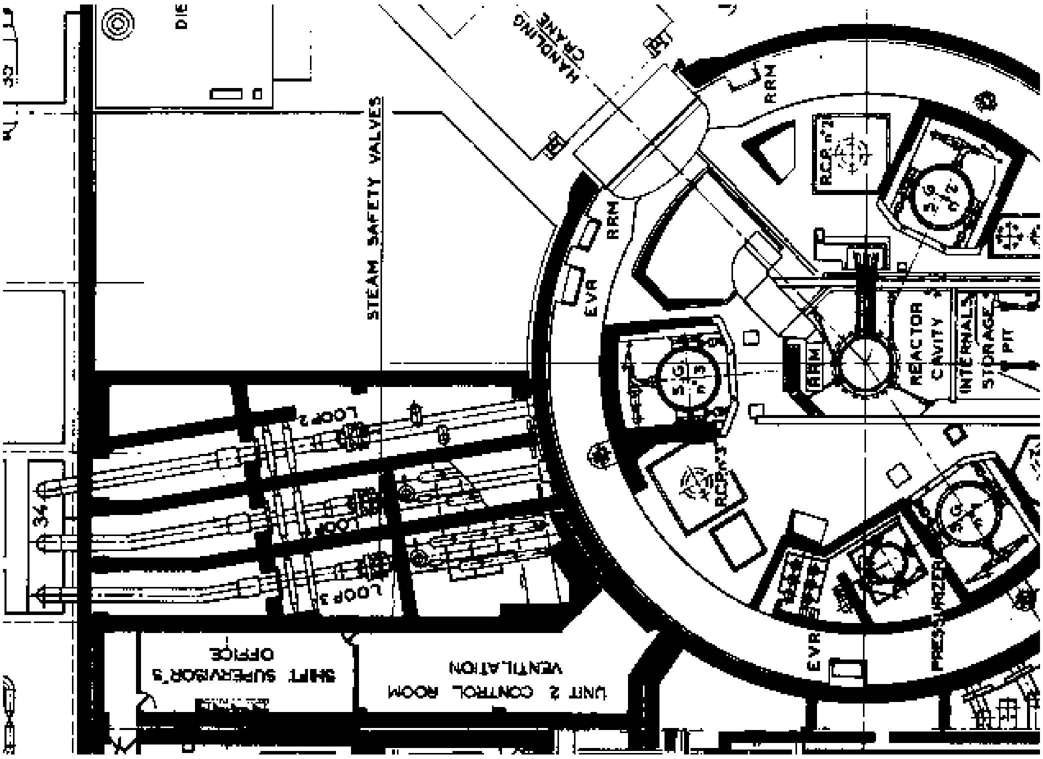
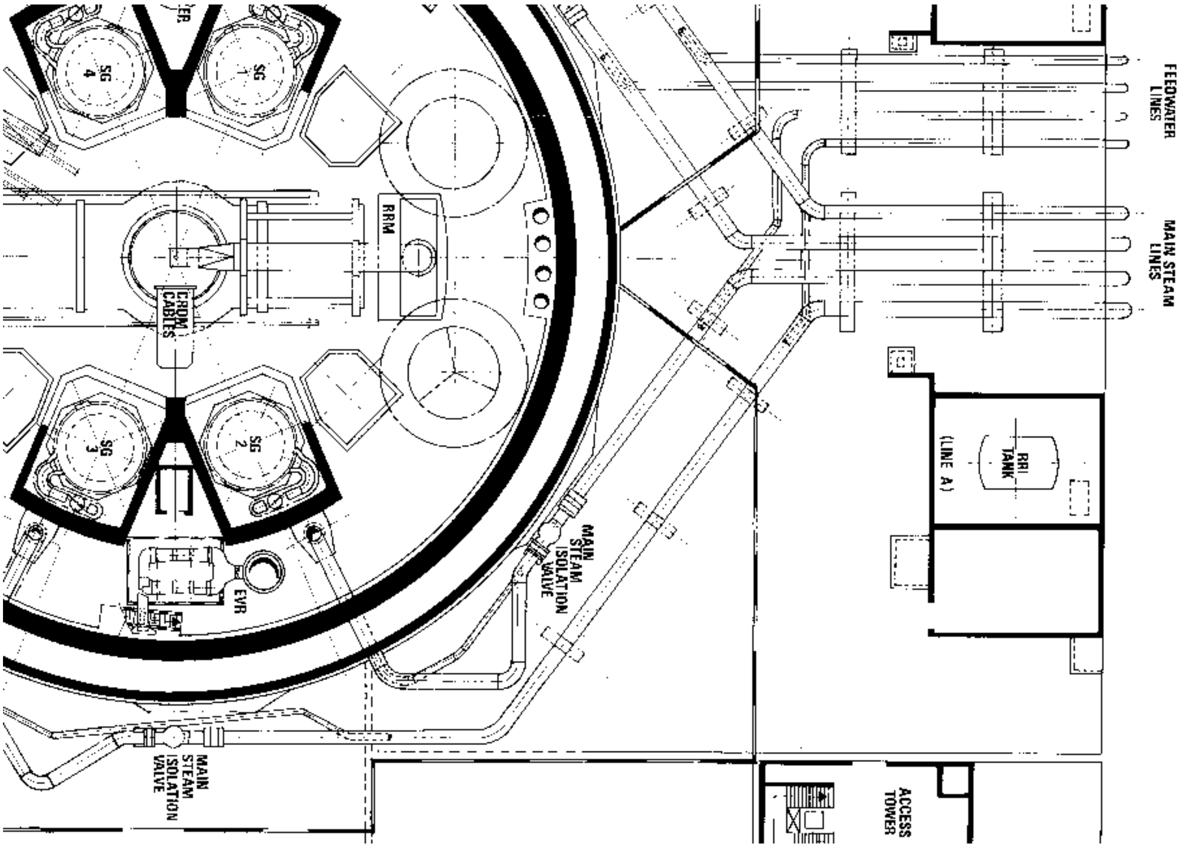


Figure II: Layout of Main Steam Lines – French 1300 MWe Model



6.7.2 Issue 10: Qualification of main steam line safety and relief valves for two-phase and water flow

6.7.2.1 State-of-the-art relevant in the Member States of the European Union

Germany

According to the German Report on the Convention of Nuclear Safety (German NSC, 1998) improvements of the steam generator feed and the main steam relief were implemented in PWRs including measures with respect to the controlled main steam relief concerned, in particular, the plant-specific technical improvements of the main steam relief control valves.

Bleed and feed capability, and hence qualification for two-phase flow, is of particular importance for main steam line relief valves in German PWRs since secondary-side bleed and feed is the first preference as an accident management measure (Smith, 1999).

The Position Report of the German Nuclear Safety Standards Commission (KTA, 1996) states that all German operating PWRs, except Biblis A and B, have fully implemented secondary feed accident management option. Biblis A and B have applied for the license.

France

N-4 (newest French series: 2 units at Civeaux and 2 at Chooz): (Berger, 1996)

Moreover, to limit the risk of water discharge through safety valves, a redundancy of the atmospheric dump system is implemented; each steam generator is equipped with two redundant safety-grade lines, each having a discharge valve and an isolation valve qualified for water discharge.

European Pressurised Water Reactor (EPR):

For the EPR it is planned to install an isolation valve on the steam line upstream of the steam relief valve.

Remark: For the general requirements on the completeness of EQ documentation see issue 19.

6.7.2.2 Situation at NPP Temelin

Issue Clarification:

The lack of qualification of main steam line safety (MSSV) and relief valves (BRU-A) for (high velocity) two phase water flow and water flow is a generic safety issue for all WWER-1000/320 reactors, as well as a specific issue applicable to Temelín. The issue was identified in the IAEA issue book (IAEA, 1996a) and measures were included in the modernisation programs for K2/R4 (IAEA, 1996d; IAEA, 1997), Kozloduy 5&6 (IAEA, 1995), etc.

In the WWER-1000/320 design, there is one BRU-A and two MSSV on each of the four main steam lines, installed upstream of the main steam isolation valve (MSIV) in the area of the intermediate building outside of the containment. Both, the BRU-A and the MSSV, can not be isolated should they open and fail to close.

In case of a primary to secondary leak (PRISE), the primary circuit water can quickly fill the SG and the steam line up to the BRU-A, which initiates the opening of BRU-A and/or of the MSSV. As long as primary coolant is being discharged to the steam generators, any time the BRU-A and or MSSV are open, the containment is bypassed discharging primary water radioactivity directly into the environment. Due to the lack of qualification of BRU-A and MSSV to operate with water or a dynamic water-steam mixtures, as it is likely under accidental conditions, they can fail to re-close after opening which results in an ongoing uncontrolled containment bypass leakage into the environment.

According to the Temelin PSA (IAEA, 1996b; Mlady, 1999) PRISE events are the most important initiators for core damage accidents at Temelin Unit 1. In case of a core damage accident involving a PRISE event, the ability to reclose BRU-A and the MSSV to avoid direct and ongoing containment bypass allows retention of some of the released fission products in the SG and the steamline. This is essential in reducing the amount of radionuclides released into the atmosphere (source term for the accident).

Proposed Upgrading Measures by the Issue Book:

The measures proposed by the member states and recommendations of the IAEA in IAEA issue book (IAEA, 1996a) (*remarks of the authors were added*):

- Qualify the safety and relief valves for water flow. (author's remark: *and (dynamic) two phase flow*)

- Study the possibilities for installing an isolating valve on the steam line upstream of BRU-A valve in order to guarantee steam generator isolation. (author's remark: *which should be qualified to work under two-phase flow conditions to definitively avoid possible damage caused by water hammer effects)*

- Effective accident management measures to reduce the consequences should be considered.

- As a long term measure, a design modification should be planned that provides adequate steam and water relief without a need to open a valve directly to the atmosphere, which can not be isolated. (author's remark: *obviously to be connected to a steam dumping device*)

IAEA Report on Temelin (IAEA, 1996c) made the following conclusions and recommendations:

- In addition, it is recommended that the Temelin NPP still investigates the possibilities for installing an isolation valve upstream of the steam dump valve to atmosphere (BRU-A). This valve should be qualified to close during water flow, so that a complete steam generator isolation can be ensured.

Results of the Workshop Rez on Issue 10, Feb. 26 - 27, 2001:

According to information presented at a workshop on issue 10 (Workshop Rez, 2001) tests for the Mochovce BRU-A type 936-150/250 were performed at the EdF-CUMULUS test facility and tests for a prototype WWER-1000 MSSV (type 1048-250/400) by Siemens KWU and Chekhov, the main designer of the valves. Referring to the ASME code, QME-1, 1994: Qualification of active mechanical equipment used in NPP, section QV (qualification of valves) on design similarities, Rez concluded that the Temelin BRU-A (type 1115-300/350) are qualified for two-phase and water flow. The similarity of Temelin MSSV (type 969-250/300) with the one tested by Siemens/Chekhov cooperation were assessed by experts and therefore the Temelin MSSV are qualified for two-phase and water flow.

Questions related to the definition of the bounding condition of the qualification tests were raised by the Austrian delegation. According to the minutes of a workshop on this issue (Workshop Rez, 2001) "*the NRI stated that selection of the bounding condition for the qualification was certainly based on long term experience of all involved parties - Framatome, Siemens and Chekhov manufacturer. Due to the fact that all the test data as well as testing philosophy is confidential property of above consortium, the Temelin NPP as well as NRI could only quote this bounding principles. However if the Austrian experts would like to know more details, they have to contact the Consortium directly in order to receive this more detail information".*

This is in contradiction with requirements that a complete set of equipment qualification documentation has to be available at the plant (for the general requirements on EQ documentation see issue 19).

Additionally the following issues were not or only partly answered in the workshop:

- Design modifications and studies on design modifications implemented or planned (in particular on the possibility to install a qualified isolation valve upstream BRU-A)

- To what extent were the results of others tests for WWER-1000 or WWER-440 valves taken into account

- Qualification requirements under severe accidents

6.7.2.3 Possible radiological impact on Austria

According to the Temelin PSA the ability of the safety and the relief valve to re-close is essential to reduce the atmospheric release for the most frequent severe accident sequences at Temelin. These SGTR sequences (including SG single tube, multiple tube, and SG collector head lift) with stuck open relief valves have no mitigative mechanism in the containment. Therefore according to the PSA level 2 for Temelin large radioactive releases into the environment would be predicted.

There exists a considerable probability that large releases of radioactivity will have an impact on Austria. The hypothetical negative transboundary impact of an accident with large release of radioactivity regarding dispersion and transport only of Cs-137 for selected historical weather situations was evaluated in the Report of the Austrian Government on the Temelin EIA No. 2 (UBA, 2000).

6.7.2.4 Important documents/information not provided

Qualification of BRU-A and MSSV:

1 Complete qualification documentation on the Temelin BRU-A and the MSSV *(requested at the Workshop Rez, 2001 on issue 10)*, including:

- Pasport, zapomo-drosselnyj klapan DU 300/350 1115-300/350-EU (rusky), Atomenergoexport, 1991.
- Pasport, klapan glavnyj, predochranitelnyj DU 250/300 969-250/300-OE01U3 PS (rusky), Atomenergoexport, 1991.

2 In particular, in accordance with the ASME code, QME-1, 1994, to document the qualification test results and the design similarities the following documentation has to be prepared *(requested at the Workshop Rez, 2001 on issue 10)*:

a) the qualification plan for parent valves (QV-8200),
b) the functional qualification report for parent valves (QV-8310),
c) the application report of the Temelin candidate valves (QV-8320).

3 Qualification reports of valve BRU-A 936-150/250 for NPP Mochovce tested at EdF test facility CUMULUS. (*Letter to EdF, see Attachment 2)*

4 Report on accident spectrum (including break spectra) covered by the test conditions of the qualification tests. *(requested at the Workshop Rez, 2001 on issue 10)*:

5 Information on other tests performed for WWER-1000 and WWER-440 MSSV and BRU-A. *(requested at the Workshop Rez, 2001 on issue 10)*:

6 Studies on qualification of BRU-A and MSSV under severe accidental conditions (resistance against hot gases with fission products, see also IAEA report on Temelin (IAEA, 1996c)) *(requested at the Workshop Rez, 2001 on issue 10)*:

Design modifications:

1 Studies on design modifications implemented or planned (in particular on qualified isolations valve upstream BRU-A) *(requested at the Workshop Rez, 2001 on issue 10)*:

2 Information on design modifications implemented in other WWER-1000 plants.

6.7.2.5 Overview of other documentation on this issue

IAEA Working Material: Safety Aspects of WWER-1000 reactors (IAEA, 1992): The main steam relief valves (BRU-A) do not have isolation valves in front of them, which is different from most western designs.

Recommendation:
Provide isolation valves in front of the BRU-A.

GRS Safety Evaluation for Stendal (GRS, 1993): GRS stated that as a result of the accident analyses performed for Stendal NPP the steam line release valves (BRU-A) have to be designed for two-phase flow. Additionally it was recommended to install an isolation valve upstream of BRU-A.

IAEA Issue Book (IAEA, 1996a) proposed the following upgrading measures:

- Qualify the safety and relief valves for water flow.
- Study the possibilities for installing an isolating valve on the steam line upstream of BRU-A valve in order to guarantee steam generator isolation.
- Effective accident management measures to reduce the consequences should be considered.
- As a long term measure, a design modification should be planned that provides adequate steam and water relief without a need to open a valve directly to the atmosphere, which can not be isolated.

IAEA Report on Temelin (IAEA, 1996c) made the following conclusions and recommendations:

The safety issue has been addressed. However, for the time being, the intent of the recommendation is only partly met.
The new designed valves have to be qualified and installed at the plant. In addition, it is recommended that the Temelin NPP still investigates the possibilities for installing an isolation valve upstream of the steam dump valve to atmosphere (BRU-A). This valve should be qualified to close during water flow, so that a complete steam generator isolation can be ensured.

Moreover, a review of the studies of the design and beyond design base accidents may lead to additional qualification being required for the steam generator safety and relief valves.

WENRA Report (WENRA, 2000): Plant-specific safety demonstration for the functioning of the main steam relief valves and the main steam safety valves under dynamic loading with a steam-water mixture still has to be fully verified. This action is underway. This function is needed to control specific primary to secondary leaks.

Czech Nuclear Research Institute Report (NRI, 2000): At experimental facility CUMULUS in France functional tests of flowing fluid change with the relief valve BRU-A type 936-150/250 EU were performed at nominal operating conditions. According to NRI test results were satisfactory and the electrical actuator showed a sufficient margin of the operational moment in all regimes. In Temelín, a geometrically similar relief valve BRU-A type 1115-300/350 EU is installed, made of the same material and with the same type of control as the tested one. "*It may be thus realistically assumed that the relief valve installed in the Temelín NPP will perform satisfactory*." The qualification program of SG safety system (2 safety valves, 1 relief valve) is carried out; within the scope of this program also the qualification of safety valves is included.

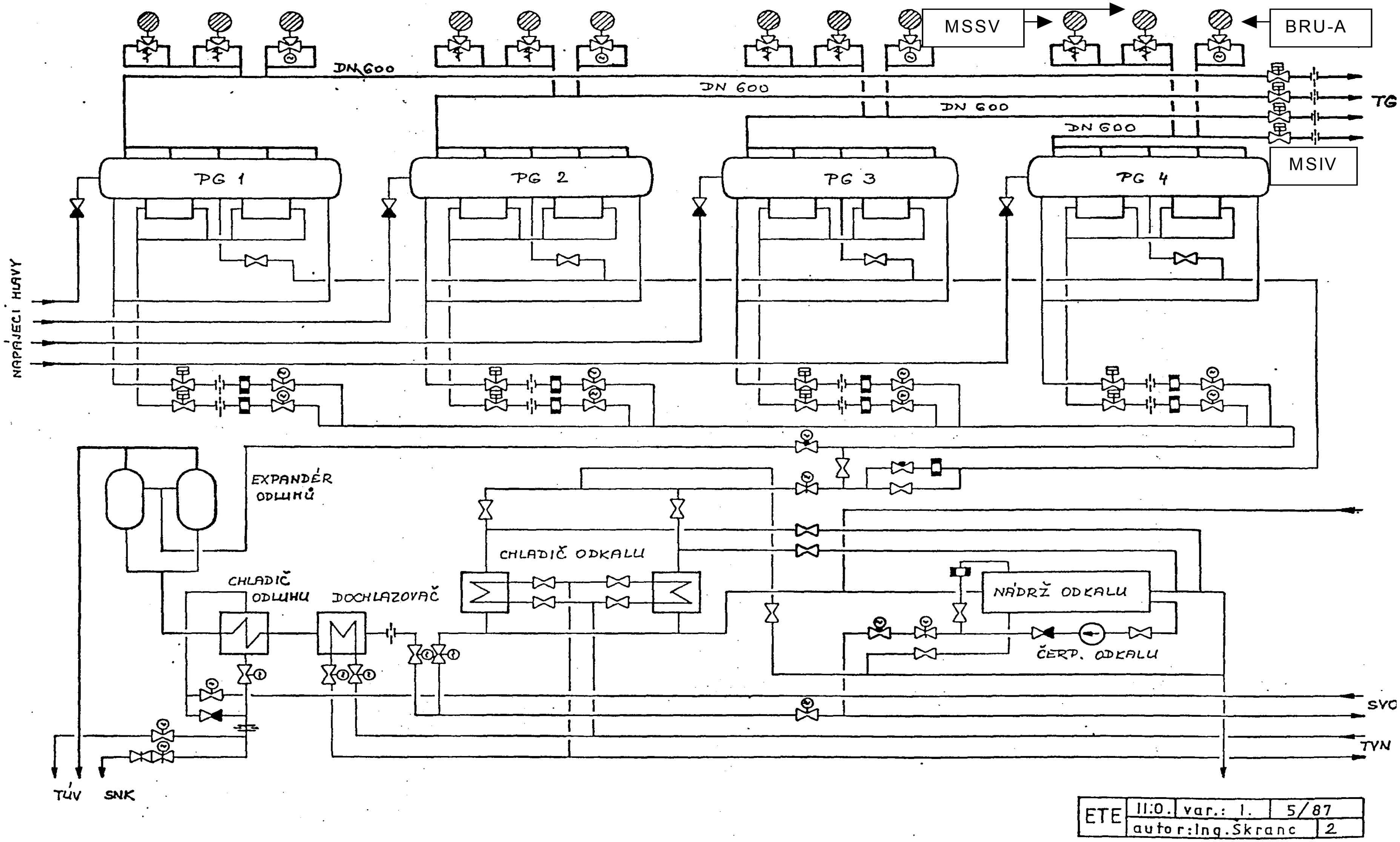
GRS Temelin Report Summary (GRS, 2000):
Part 1 - Summary of the results of the evaluation of selected safety issues for the Temelin NPP status - 15.8.2000 states that no plant-specific proof for the planned functioning of the BRU-A and the steam line safety valves has been given.

Part 2 - Extension of part 1, taking into account additional SUJB information: Based on the information provided by SUJB on tests at CUMULUS the statement that BRU-A for Temelin is qualified is plausible. However GRS could not assess the tests.

Temelin PSA and POSAR: Neither Temelin PSA from 1995 nor Temelin POSAR provides any information on the designer, qualification tests or other measures to solve this safety issue.

PRISE in WWER reactors (IAEA, 2000):
The IAEA document on PRISE in WWER reactors from 2000, states that at WWER reactors BRU-As and SGSV are originally qualified for steam flow only and not for water-steam mixtures. For such a qualification, it is likely (under SG overfill conditions) that once these valves open, they can remain fully or partially stuck open.

6.8 Attachment 2: Arrangement of MSSV and BRU-A valves on steam-lines



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)

Table of Contents

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 m^3 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 m^3 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 m^3 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 <u>not</u> 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 <u>before</u> 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 <u>clearly</u> 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 <u>before</u> 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 6.6×10^{-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 (eddie-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, <u>however the design analysed in the PSA includes certain planned modifications which were credited in the analysis</u>. These credited modifications <u>include</u> those related to the SG collector restraining bolts and implementation of symptom-oriented EOPs. These modifications are <u>fully</u> 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 <u>excluded</u> from the PSA based on design changes. Thus, the PSA shows <u>no</u> CDF contribution from SG collector failures. The contribution to CDF comes from header <u>leakage</u> (not failure), and <u>fully</u> 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 <u>already</u> 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 1.3×10^{-4} per year. This would reduce the SGCL CDF value from 4.33×10^{-5} per year to 2.17×10^{-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 2.1×10^{-2} per year. This would reduce the SGTR CDF from 2.27×10^{-5} per year to 1.08×10^{-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 2.36×10^{-5} per year to 1.99×10^{-5} per year. With these three corrections, the internal events CDF would drop from 8.96×10^{-5} per year to 5.24×10^{-5}per year. Containment bypass sequences (SGCL plus SGTR) would still contribute 3.25×10^{-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 1×10^{-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 5×10^{-5}, 3×10^{-5}, and 2×10^{-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 1×10^{-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.

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8 Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs) - Issue 06

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8.1 Introduction

Symptom-oriented emergency operating procedures (SEOPs) are designed to guide and support the operating personnel in accident prevention. Severe accident management guidelines (SAMGs) extend the SEOP concept into mitigation and/or limitation of accident consequences such as core melting.

The implementation of SAMGs or equivalent procedures as a prerequisite for adequate Defence in Depth provisions in case of accident is the state-of-the-art in Member States of the European Union.

8.2 Identified Problems

SAMGs are of special importance in the case of the Temelin prototype plant and have been recognised a necessity by CEZ. However, SAMGs have not yet been developed and implemented in the Temelin Defence in Depth Concept.

SAMGs have also to be taken into account when developing EOPs in order to achieve consistency for appropriate decision making and subsequent actions. Transitions between the EOPs and SAMGs must be defined. Careful harmonisation and integration of SAMGs into EOPs should therefore precede the verification and validation process needed for proper implementation and possible use of both. In view of the delay in supplying and implementing SAMGs, the symptom based EOPs as prepared and validated for Temelin Unit 1 must be considered tentative at this stage of development.

Given the risk implication to Austria, its population and environment, when one of the Levels of Defense - here Level 4 , the last on-site, the containment - is challenged in a severe accident, the implemented EOPs and SAMGs measures are of paramount importance.

It should be recalled that in all accident comparable conditions, when SAMGs are applied the Defense in Depth layers Level 1 up to Level 3 are at the verge of failure or have failed already. The SAMGs should provide for maintaining Level 4 of the Defense in Depth as an ultimate resort before large releases of radioactive material into the environment takes place.

Furthermore, for accidents with containment failure or bypass, since the Defense in Depth layers Level 1 up to Level 4 have failed already, SAMGs are part of Level 5 in that they are supposed to provide for actions to limit or reduce otherwise large releases of radioactive material into the environment to the minimum achievable under severe accident conditions and therefore support emergency management and offsite emergency response.

For these reasons the SAMGs must be available to the operator and the emergency support staff to be able to prevent, to mitigate and if necessary to limit severe accidents while following ALARA requirements in the best way achievable.

Emergency management and response depends strongly on the proper use of the lead time available for intervention. International Emergency Exercises (like INEX) have provided the participating organizations with an appropriate picture of the information flow and flow delays to be expected.

To the extent necessary to ensure optimal management strategies, the emergency management of possibly affected countries should also be informed of the actions foreseen in the SAMGs. Should the need arise, additional explanations by the Operator could be asked and given to prevent misunderstandings and misinterpretations resulting from inappropriate use of the information given.

8.3 Solutions to Identified problems

- Remaining EOPs and the complementing SAMGs, as well as the transition rules should be implemented and updated for NPP Temelin as the results of the analyses of severe accidents involving containment failure (see Chapter 7) become available.
- The EOPs and SAMGs should be made available to Austria to the extent necessary to optimise emergency planning in case of beyond design base accident scenarios with potential crucial consequences to Austria, its population and environment.

Timeline: Neither for Temelin Unit 1 nor Unit 2 would the state-of-the-art relevant in Member States of the European Union permit operation or even fuel loading before resolution of this issue. All the necessary analyses and related measures could be completed within one year.

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

Category: high safety significance/large or moderate deviation
Levels of Defence in Depth are considerably diminished.
(The high safety significance is because SAMGs are a relevant support to Defence in Depth (DiD) and provide the potential to reduce severe accident consequences.)

8.4.1 State-of-the-art

There is only a very limited number of NPPs that was taken into operation during the last decade or after the 1996 SAMG voluntary program in the USA became effective. It is therefore obvious that a derived European practice must rely on requirements formulated during this time rather than practical examples of licensing practice.

Extensive information generated internationally regarding the initiation and propagation of severe accidents has permitted the characterisation of residual risks and the identification of plant upgrades that have a potential for significantly reducing such risks. Of particular importance are the development of a set of accident scenarios and their occurrence frequencies, the mode and timing of containment failure relative to reactor vessel failure, the projected consequences and the emergency response measures. This has led the US NRC to launch a plan to address severe accidents for operating nuclear power reactors in 1985.

In 1991 the recommendations of the IAEA Conference on *"The Safety of Nuclear Power: Strategy for the Future"* identified the need for accident management and core damage mitigation (see Issue 1 Topic 5: *'Fundamental Principles for the Safe Use of Nuclear Power; The Safety Objectives and their Application for Accident Prevention and Mitigation: Limiting the consequences of severe accidents')*.

Symptom-oriented EOPs were adopted throughout the EU following the 1979 *Three Mile Island* accident and the 1986 *Chernobyl Unit 4* accident. Severe accident management guidelines (SAMGs) extend the EOP concept from accident prevention into accident consequences mitigation. EOPs and SAMGs must consequently allow for co-ordinated sequences of actions, particularly when entered and exited.

A distinguished group of senior regulators of 23 member states of the IAEA (including the Czech Republic) took an unprecedented strong position advocating SAMGs introduction in 1996 when stating in the conclusions of their discussions /Ref. 8/: "*There was unanimity on the need for the operating organisation to have emergency plans and the requirement for their acceptance or approval by the regulator. A general view was taken that the issue of severe accident management was an important development which warranted the operating organisation paying special attention to measures to prevent severe accidents and also to consider the way events should be controlled following the occurrence of a severe accident."*

In Europe SAMGs were introduced in Belgium, The Netherlands, Spain, Slovenia, Switzerland and Finland. Germany, France, Sweden and the U.K. have taken diverse approaches in adopting accident management concepts, some of them already in the early 1980's. Work is still going on in a number of countries in Europe such as in Hungary, Bulgaria, Slovakia, Czech Republic, Lithuania, Ukraine, Armenia, Russia.

It is standard Westinghouse practice for all of its NPPs to implement both symptom-oriented EOPs and SAMGs. The Westinghouse Owners Group (WOG) has supported the original development. Practically all operators of PWRs, among them several utilities operating WWER-1000 reactors have prepared the documents, procedures, flow charts, plant status trees as required, on the basis of then-available information. Updated and optimised procedures and guidelines were and are being prepared for the operators as additional information becomes available.

SAMGs together with INSAG-10 Defense in Depth - Level 5 provisions establish appropriate information links and the transition to the off-site emergency management operations vital to a comprehensive ALARA concept implementation. This information is of vital interest to all those involved in Accident Management as well as in Emergency Preparedness within the Czech Republic and abroad.

8.4.2 Situation at NPP Temelin

Symptom-oriented EOPs are being implemented at Temelín Unit 1, but SAMGs have not been implemented yet. Thus, Defence in Depth cannot be considered adequate when compared to the extent required in Nuclear Power Plants in Member States of the European Union.

The Technical Support Centre is equipped and manned to provide support to the operator in case of a Severe Accident. It is supposed to make extensive use of the SAMGs. The training and experience of the personnel are helpful and necessary prerequisites for Severe Accident Management, but they cannot replace the SAMGs.

8.4.3 Deviation from the state-of-the-art

The delay in preparation and implementation of SAMGs and of their integration with the implemented EOPs beyond the start-up of the plant is a deviation from the state-of-the-art in Member States of the European Union.

It is understood that SAMGs for Temelin are being developed at this time. The view taken by the OSART mission, that the operator is in the right way of having all this tools - not "completely" but – “well” implemented must be seen against this background.

8.5 Technical Arguments

8.5.1 Defense in Depth

"Defence in Depth In Nuclear Safety", INSAG-10, issued in 1996 by the IAEA, Chapter *Defence in Depth (DID) - '**Level 4**: Control of severe conditions including prevention of accident progression and mitigation of the consequences of a severe accident'*, paragraph 45, states the following:

"...Essential objectives of accident management are:

1. *to monitor the main characteristics of plant status;*
2. *to control core subcriticality;*
3. *to restore the heat removal from the core and maintain long term core cooling;*
4. *to protect the integrity of the containment by ensuring heat removal and preventing dangerous loads on the containment in the event of severe core damage or further accident progression;*
5. *to regain control of the plant if possible and, if degradation cannot be stopped, delaying further plant deterioration and implementing on-site and off-site emergency response…."*

The Defence in Depth (DID) objectives 4 and 5 and partially even 3 have not been met adequately yet for this composite NPP.

The support to the Control Room as well as the knowledge basis at the Technical Support Centre (TSC) rely on the accident analyses results for the efficient accomplishment of the actions required to provide for, or even re-establish, a sufficient level of Defence in Depth. Conclusions to be drawn for SAM from this knowledge are left entirely for the emergency management staff requiring sufficient lead time before any action is taken (see the Appendix for the full appreciation of the preparatory work SAMGs can save). SAMGs are adopted to reduce significantly the lead time requirements.

It is understood that DID **Level 5** of INSAG-10: "*Mitigation of the radiological consequences of significant external releases*" also requires the establishing of SAMGs.
From the above, the following conclusion can be drawn: Defence in Depth - a crucial element to the ALARA concept - is not implemented to the extent attained in most European Nuclear Power Plants.

8.5.2 Timelines and safety culture enhancement

The delay of the introduction of SAMGs at the Temelin NPP is surprising for a number of reasons:

- Undisputedly, the need for the introduction of SAMGs has been clear since the time of the decision to finalise the plant;
- WWER-1000 Temelín NPP is a prototype reactor for which extensive design and accident analyses were required. Therefore a well established understanding of accident sequences in the DBA and the BDBA ranges should and could have been created;
- The results from the PSAs conducted have indicated the special need for SAMGs, especially with regard to severe accident sequences involving containment bypass and failure
- The WWERs owners group has no working program for WWER-1000 Temelín units;
- The decision basis actually available for the TSC (Technical Support Centre) personnel has not been validated, nor was the implementation subject to an independent audit; BDBAs are to be treated on a case by case basis.
- The original designer developed his own line of SAM policy. Results from this work are not known to have contributed to Temelin's SAM policy.

All these facts should have led the operator to devote considerable effort to the SAMGs development. However the development to date was limited to the implementation of EOPs and to the further analyses of severe accident occurrences and phenomena, all of which are prerequisites, but do not substitute fully developed SAMGs. These factors together are usually taken as symptoms for considerable limitations regarding the type of safety culture required for the operation of NPPs.

8.5.3 Collateral requirements

Availability of prerequisites — Severe accident studies [1] and the Temelin PSA results are available and can be transformed into SAMGs in the short term. The results of the additional severe accident analyses in view of possible containment failure as described in the Cluster "Containment" (Chapter 7) can be integrated as they become available. Insights from PSA should be used to establish organisational and procedural measures for operators and Technical Support staff supposed to cope with evolving severe accidents.

Implementation — Verification and Validation will have to be performed using the Temelin full scope plant simulator. Documentation and Training material will have to be produced, training courses will have to be generated, trainers, plant staff and technical support staff will have to be to be trained. Additional computational support will have to be established and implemented at the TSC (Technical Support Center).

Maintenance — SAMGs as developed now will be limited to adaptation of accident sequences to current plant status. Updates will have to be made according to quality assurance (QA) and implementation procedures similar to those adopted for EOPs. Computational support, maintenance, quality assurance QA and implementation procedures for SAMGs are similar to those for EOPs as well.

Audits — Self-assessment as well as independent audits are to be foreseen for applications suitability, training efficiency and computing support validity. Status verification procedures are to be applied. A frequency of not less than once every two years appears to be adequate unless plant changes require a considerably higher frequency.

8.5.4 Involvement of the licensing authority

The usual involvement of the licensing authorities is not limited only to the Design Basis approach. It also includes the auditing of the safety related commitment of operating organisation, for example the introduction of PSAs on all levels, the implementation of SAMGs, etc.

Oversight by a regulatory body whose primary purpose is safety can stimulate complete and objective analyses as well as comprehensive improvements. The licensing body should evaluate the results of the overall severe accident program and document its own safety conclusions and recommendations.

[1] Severe accident studies referenced on the NRI report from March 2000 in response to the IAEA WWER-1000 safety issues report.

Comparable activities have not been identified to the extent required in the case of the SA precautions at the Temelin NPP.

8.6 References

1. Temelín POSAR, Rev. 1, December 1999.
2. Temelín PSA, December 1995.
3. F. Hezoucky, Temelín NPP Status: The Challenge of Safety Improvements, Uranium Institute 25th Annual Symposium, London, 30 August - 1 September 2000.
4. IAEA-WWER-SC-171 (review of WWER-1000 issues resolution at Temelín).
5. IAEA-EBP-WWER-05 (WWER-1000 safety issues report).
6. Nuclear Research Institute Řež report, March 2000 (response to IAEA report).
7. Nuclear Research Institute Response to the Austrian Issues, March 2001.
8. Development of measures to assess safety of existing NPPs and the effectiveness of regulations and regulatory actions (including 'prescriptive' and 'performance based' approaches). Peer discussions on regulatory practices, IAEA Document PDRP-1, Vienna 1996

8.7 Appendix: Severe Accident Management Guidelines purpose

SAMGs provide a symptom based, structured guidance to plant staff responsible for stabilising and recovering the plant from a severe accident.

The SAMGs provide the comprehensive guidance necessary to:

- ***Diagnose plant conditions*** - a symptom based approach is employed using only measurable plant parameters;
- ***Prioritise response*** - symptom based parameters are prioritised based on the time available for response;
- ***Assess equipment availability*** - availability of equipment for response is determined (a key item in this part of the process is prioritising the recovery of equipment when it is not available);
- ***Identify and assess negative impacts*** - the negative impacts of implementing available equipment are identified next. This part of the process also includes the identification of additional actions that can mitigate the negative impacts;
- ***Determine whether to implement available equipment*** - based on a comparison of the negative impacts to the consequences of taking no action, the decision whether to implement a given strategy can be reached;
- ***Determine whether implemented actions take effect*** - after the strategy is implemented, it is necessary to know if the actions are effective and if the negative impacts are still acceptable;
- ***Identify long term concerns for implemented strategies*** - after the strategy is implemented, there may be additional long term actions required to maintain the strategy (e.g. refilling tanks).

SAMGs implementation might require changes to those EOPs that are supposed to serve for the transition between the preventive and mitigative phases of the accident management.

Such transitions from the EOPs to the SAMGs might occur in the case of:

- ATWS (Anticipated Transients without Scram) events
- Loss of core cooling
- Station blackout

The SAMGs are the tool for Technical Support Centre staff in co-operation with shift staff to stabilise and recover the plant from a severe accident. All means suitable to help this goal are of use.

9 Technical Basis for Emergency Planning - Issue 29

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9.1 Introduction

Emergency Preparedness is defined as 5th level in the Defense in Depth Concept according to IAEA. Its aim is to reduce the possible environmental impact and health consequences of radioactive releases for the population. Emergency preparedness is based on severe accident analyses for the specific plant and dose calculations for the surroundings. One important part of the emergency preparedness concept is the “in advance” determination of emergency planning zones (EPZ) and corresponding protective actions.

The necessity for emergency preparedness measures may well extend into neighbouring countries in the case where the NPP is constructed near national borders.

9.2 Identified problems

- The definition of the Emergency Planning Zones (EPZ) for Temelin NPP is based on IAEA TECDOC-953 (Czech Rep. Report 2001; SUJB April 2001; SUJB May 2001). However, while the Precautionary Action Zone (PAZ) and the Urgent Protective Action Zone (UPZ) were clearly declared, there is contradictory evidence on the determination of the Long Term Emergency Planning Zone (LPZ): While the official documents state that the LPZ was not explicitly determined (Czech Rep. Report 2001), it was claimed in the discussions that the LPZ covers the rest of the Czech Republic. It is clear that the LPZ will include extended parts of neighbouring countries such as Austria and Germany. Compliance with the recommendations of IAEA-TECDOC-953 and –955 requires an explicit determination of the LPZ also across borders.

- Czech presentations at the Post-Prague meeting (Handouts, April 2001), the documentation on severe accidents in the frame of the Temelin EIA III (SUJB, April, 2001; SUJB, May 2001) and Austrian calculations for a moderately severe accident source term indicate that intervention levels for sheltering and iodine prophylaxis can be exceeded also in Austria. This means, that urgent protective measures could be necessary in Austria in case of a severe accident. However, there are discrepancies in calculated doses, which cannot be clarified on the basis of information available at present (source terms and meteorological conditions are not sufficiently known).

- Emergency Planning is normally based on the results of the most challenging severe accident analysis performed for the nuclear installation. This is not the case for Temelin NPP, as the severe accidents analysis for NPP Temelín did not take into account in an appropriate manner some threats to containment integrity (see Cluster “Containment Integrity in Severe Accidents”) for which greater atmospheric releases could be expected than those considered at present.

- EP must aim at observing the ALARA principle requirements on the one hand and not imposing more severe measures than necessary on the other. Thus it is essential that EP be based on a realistic estimate of possible source terms. The source term fixed by the Czech side for dispersion calculations is a combination of the inventory of a German 1300 MW power plant and emission rates for three classes of radio nuclides. More precise information on source terms could be helpful in establishing more efficient Emergency Planning in Austria.

9.3 Solutions to identified problems

Within the framework of the pertinent bilateral agreement between the Czech Republic and Austria,

- information on the EPZ update as a result of more complete severe accident analysis as well as updates resulting from implemented safety upgrading etc., should be provided, and

- the co-ordination of emergency planning and enhanced co-operation in radiological protection (as offered by the Czech side during the “Post Prague Meeting”) should be accomplished.´

As severe accidents are shown to have transboundary effects,

- the LPZ should be determined - both on Czech and Austrian territory as appropriate, the transboundary approach and advance calculation being specifically recommended by IAEA-TECDOC-953, 1997 and -955, 1997.

- greater transparency of the calculational methodology and the specific input used to determine the EPZs and the expected doses would be helpful.

For independent assessment of the extent of emergency preparedness required, more precise information on source term functions (time dependent release of radiologically important radio-nuclides and their chemical forms) in absolute and relative terms as well as release location, souce height and released energy, for all severe accidents to be considered would be necessary.

9.4 Deviation from the State-of-the-Art and its Significance

9.4.1 Determination of the Long Term Planning Zone (LPZ) in advance and across boundaries

The long term emergency planning zone (LPZ) was most likely not determined for Temelín in the Czech Republic and certainly not in Austria (Czech Rep. Report, 2001, SUJB, April 2001; SUJB, May 2001 and Commission, 2001). This is not in compliance with the IAEA-TECDOC-953 and -955 in which advance determination of the LPZ and extension beyond national borders is recommended:

"It is the area where preparation of effective implementation of protective actions to reduce the long term dose from deposition and ingestion should be developed in advance". and on page 11: "*It is important to note that the* (EP) *zones do not stop at national borders."*

Within the LPZ a large amount of information on diverse matters, including legal considerations, food logistics, farming practices, possible agricultural countermeasures, potential relocation resources has to be collected in advance to develop and validate effective measures. As this involves considerable effort and costs, it will only be done within an explicitly defined LPZ. Therefore the LPZ should also be determined for neighbouring countries.

The concept of EPZ is summarised in figure 1 (IAEA-TECDOC-953, page 11).

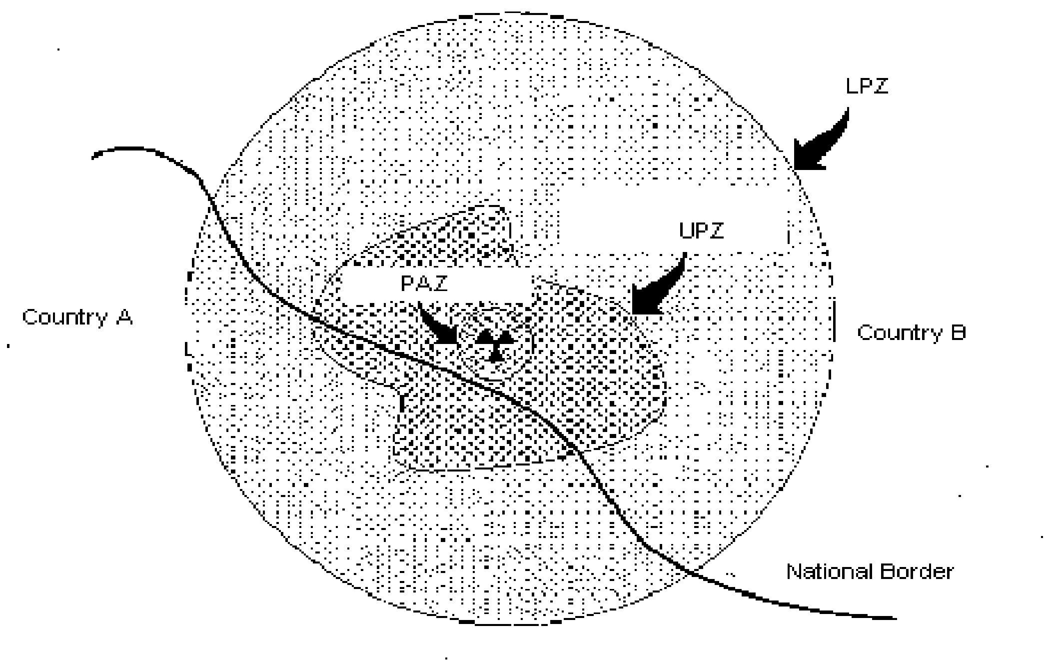

FIG. 1. Concept of emergency planning zones.

9.4.2 Advance information of the public on possible releases from severe accidents

According to the Commission Communication on the implementation of the Council Directive 89/618/EURATOM of 27 Nov. 1989 (Council Directive 89/618/EURATOM), on informing the general public about health protection measures to be applied and steps to be taken in the event of a radiological emergency (91/C 103/03), prior information on various types of radiological emergencies and their consequences for the population and the environment should be provided to the population living near an NPP including:

- *the unlikely possibility of an accident having any impact on the population,*
- *the types of emissions (gas, dust, liquid) which would be released from the installations in the event of an accident, and how far and how quickly they would spread*."

Thus the request for a Temelín specific source term in sufficient detail to calculate the resulting impacts on Austria is in line with EU standards.

9.5 Technical Arguments

9.5.1 Completeness of accident scenarios taken into account for Emergency Planning

The Decree of the Government of the Czech Republic No. 11/1999 on the Accident Planning Zone requires determination of an accident planning zone which has to be approved by the State Office for Nuclear Safety. According to this decree determination of EPZ must consider all possible radiation accidents with probability of occurrence for the particular nuclear facility higher or equal 10^{-7}/year.

According to the presentations of the Workshop on the Post-Prague issues the Temelín EPZ were determined on basis of severe accident sequences for 3 initiating events: Major leaks from primary to the secondary circuit, Large Loss of Coolant Accidents (LOCA) and Station Blackout with permanent loss of all active safety systems.

It is however unclear whether severe accidents for the reactor in the shut-down mode and due to the loss of cooling of the spent fuel pool have been taken into account for EP. The Temelin PSA indicates that the frequency of these accidents is higher than 10-7/year and thus they need to be considered. Source terms for these accidents could be rather large.

As elaborated in Chapter 3 (Cluster Containment Integrity and Severe Accidents) it is not considered proven that containment integrity is assured under the selected severe accidents conditions. The calculated release of radioactive material and the EPZ determination at present is based on the assumption of an intact containment applying leak functions close to the nominal rates. Some of these sequences could have probabilities above 10^{-7}/year. Thus the present severe accident analysis might not be sufficiently comprehensive. Conservatism requires the use of a source term corresponding to containment failure for emergency planning until either containment integrity is proven or an update with more detailed, plant specific source terms for containment failure are available.

9.5.2 Determination of EPZ

The IAEA documents provide the following criteria for the definition of EPZ (IAEA-TECDOC-953 and IAEA-TECDOC-955):

PRECAUTIONARY ACTION ZONE (PAZ) - 3-5 km: The size of the PAZ is based on the best estimate consequences in the case of a worst accident. Protective actions should be implemented for the whole zone whenever the conditions for a severe accident develop.

1. Urgent protective actions taken before or shortly after a release within this zone will significantly reduce the risk of doses above the early death threshold in the worst case (considering potential releases and meteorological conditions).
2. Urgent protective actions taken before or shortly after a release within this zone will prevent dose above the death thresholds for most severe accidents at this facility
3. For an atmospheric release under average meteorological conditions, this zone covers distances where about 90% of the off-site risk of serious deterministic health effects could occur.

URGENT PROTECTIVE ACTION ZONE (UPZ) - 10-25 km: The UPZ is the area where preparations are made to promptly perform environmental monitoring and implement urgent protective measures based on the results. Plans and capabilities should be developed to implement sheltering or evacuation and distribute thyroid blocking agents (if appropriate).

1. Urgent actions must be taken within 4-12 hours within the zone to significantly reduce the risk of doses above early death threshold for the worst case severe accidents (considering potential releases and meteorological conditions).
2. This distance provides approx. a factor 10 reduction in concentration (and thus risk) compared with the PAZ.
3. deterministic health effects could occur.
4. Detailed planning within this zone provides a substantial base for expansion of the response efforts in the event of the worst severe accidents.

LONGER TERM PROTECTIVE ACTION ZONE (LPZ) – 50-100 km: Area where preparations for effective implementation of protective actions reduce the risk of deterministic and stochastic effects from long term exposure due to deposition and ingestion of locally grown food should be developed in advance:

1. Doses from ground contamination warranting relocation are unlikely beyond this distance for most accidents.
2. This distance provides approx. a factor of 10 reduction in concentration (and thus risk) compared to UPZ.
3. LPZ covers distances where about 99% of the off-site risk of dose above the GILs (generic intervention levels) could occur.

While the PAZ and the UPZ were determined for the Temelín NPP, the LPZ was not determined for Temelín according to the Czech document prepared mid march for the trilateral meetings (Czech Rep. Report 2001) and the summary papers on severe accident published within the Temelín Environmental Impact Assessment (EIA) III (SUJB, April 2001; SUJB, May 2001)

On the other hand it was stated at the Post-Prague meeting that all the territory of the Czech Republic is LPZ and that in case of a radiation accident, depending on its development and extent, the long-term protective measures will be realised throughout the country, based on radiation monitoring results.

The IAEA-TECDOC-953 and 955 suggests LPZ to 50-100 km distances for NPPs. Therefore parts of the Austrian territory would lie within the Temelín LPZ (minimum distance from Temelin to the Austrian borderline is at about 50 km). Which means that in this area preparation of effective implementation of protective actions to reduce the long term dose from deposition and ingestion should be developed in advance. The IAEA TECDOC-953 also states (page 11): "*It is important to note that the zones do not stop at national borders."*

The International Commission for Independent Safety Analysis (ICISA) of NPP Krsko established by the Slovenian Government on April 1992 with the participation of representatives from the neighbouring countries, among them Austria and Italy, in its Final Report, (ICISA 1993), (Chapter 4: Emergency Planning and Preparedness) made the following recommendations regarding the transboundary character of EPZ for Krško NPP:

"The 10 km EPZ is small enough to only require consideration of Slovenia, and not Croatia when planning for Emergency actions (such as evacuation). The EPZ includes about 37,000 people in Slovenia.

In the plan, the 25 km Ingestion Pathway Zone shows population figures for Slovenia but is blank for Croatia, even though about 20% of the total zone is in Croatia. Population effected in the Slovenian Ingestion Zone was estimated at about 110,000 people. Population potentially impacted in Croatia was estimated to total about 80,000.

Considering the general necessity for mutual assistance of neighbouring countries in case of a nuclear emergency and the limited extension of the Slovenian national territory ICISA recommends to the Slovenian government to establish "ad hoc" agreements with the neighbouring countries for exchange of information and to plan the mutual assistance that should be activated in case of a nuclear emergency (technical support and materials) in advance."

Possible transboundary impact of Temelín is not elaborated in any document and was only addressed in an aside during the Post-Prague meeting. A calculation for the extent of the LPZ on the Austrian territory has not been provided.

9.5.3 Source Terms for Severe Accidents

Information on the release fractions for the V- and the AB_01 severe accident scenarios and an inventory said to be appropriate for Temelín has been provided in the EIA process (SUJB, May 2001):

- Fractional cumulative release of the **V-sequence**: The most important leakage of FP is occurs during in-vessel phase of accident (from 35th to 90th minute): Noble gases : ~0.78, Aerosols-volatile FP : < 0.19, Aerosols-non-volatile FP: < 0.01

- Total cumulative leakage expressed as a fraction of core initial inventory for the **AB-sequence (AB_01)**: Noble gases: <4.0E-3, Aerosols-volatile FP: < 8.0E-5, Aerosols-non-volatile : from 2.0E-6 up to 6.0E-5.

- The radioactive inventory of a typical German 1300MWe PWR

The release fractions are given in less detail than usual (just 3 classes of nuclides) and the inventory is not reactor-specific. There is no further information on source term functions, e.g. time dependence of release of radiologically important radio-nuclides, the species and their chemical forms or release height and released energy.

A detailed description of the accident scenarios leading to these source terms, including containment behaviour under these severe accidents, was not made available. No source terms or accident scenario descriptions were made available for the ST1-ST5 cases (see Table 1).

This makes it difficult to assess the possible transboundary effects of severe accidents. As discussed above a complete severe accident analysis could lead to much larger source terms.

9.5.4 Dose calculations and region of Urgent Protective Measures

Calculations presented at the Post-Prague workshop (Handouts, 2001; SUJB, April 2001) indicate that the intervention levels for urgent protective measures could be reached also on the Austrian territory (minimum distance: ~50 km). The same results are presented in some more detail in the documentation "Principles and methods of emergency planning and response at NPP Temelín including assessment of beyond design and severe accidents consequences" (SUJB, April 2001). A large steam generator by-pass sequence ("V-sequence" in Table 1) would – under very stable conditions (stability class F) without rain - lead to a two days dose of 10 mSv for distances larger than 40 km. The lower intervention limit for sheltering and Iodine prophylaxis is 5 mSv (Handouts, 2001; SUJB, April 2001).

For a dry containment by-pass release after core melt with leak rates of 100% per day up to 100%/hr the IAEA TECDOC-955 (page 101) estimates that urgent protective action distance for thyroid blocking and shelter is at least 50 km under dry average weather conditions.

Calculations with the PC-Cosyma code based on the V-sequence source term lead to doses beyond the intervention levels for Urgent Protective Measures under neutral as well as stable conditions (stability classes C and F) up to distances well above 100 km[1]. The significant discrepancies in the results indicate, that the source term in the Czech calculations was probably much smaller than that made available in the EIA.

The results also demonstrate that the meteorological conditions on which the Czech calculations are based are not conservative – even with the simplified homogeneous meteorological conditions the stable case with light precipitation (i.e. precipitation everywhere) results in higher doses than the dry case (Table 2). For larger distances this

[1] Calculations for greater distances should not be made with a Gauss-type model as used in the PC-Cosyma code; they require a different model approach.

would be even more pronounced if precipitation set in only at some distance from the source.

In any case, the results of all the calculations demonstrate that in case of this specific severe accident scenario it is likely that Urgent Protective Measures (at least sheltering and Iodine prophylaxis) would have to be implemented on Austrian territory. Of the larger cities for instance Linz (211,000 inhabitants) and Passau (51,000 inhabitants) could be affected.

Summarising, it is clear that significant parts of Austria lie within the LPZ. This implies the possible necessity of relocations. In some parts of the LPZ, which could stretch as far as Linz or Passau, urgent protective measures, including sheltering and iodine prophylaxis, could also be necessary.

Table 1: Distances up to which intervention levels are reached for different accident sequences for dry conditions.

Stability class F						
Sequence	2 days			7 days		
	Intervention Level			Intervention Level		
	5 mSv	10 mSv	50 mSv	50 mSv	100 mSv	500 mSv
AB_01	8 km	5 km	-	1 km	-	-
AB_02	14 km	8 km	2 km	2 km	1 km	-
AB_03	18 km	11 km	3 km	4 km	2 km	-
AB_04	16 km	9 km	1 km	2 km	-	-
ST_V	>40 km	>40 km	-	-	-	-
ST 1*	35 km	23 km	2 km	3 km	2 km	2 km
ST 1**	35 km	17 km	5 km	5 km	3 km	2 km
ST 2	-	-	-	-	-	-
ST 3*	27 km	19 km	2 km	2 km	2 km	-
ST 3**	21 km	14 km	2 km	3 km	2 km	-
ST 4	-	-	-	-	-	-
ST 5	5 km	2 km	-	-	-	-

Note:
ST 1* and ST 3* – calculations for real terrain, direction Týn nad Vltavou
ST 1** and ST 3** –calculations for real terrain, direction České Budějovice

Table 2: Two days effective dose for the plume centreline for the V-sequence (SUJB, May 2001) for different meteorological conditions calculated by the PC-Cosyma code

	Two days effective dose (mSv)	
	Distance: 60 km	Distance: 100 km
Neutral conditions (D)	90	50
Stable conditions (F)	200	100
Neutral conditions (D) with low intensity precipitation	300	~140

9.6 References

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Council Directive 89/618/EURATOM	Council Directive 89/618/EURATOM on informing the general public about health protection measures to be applied and steps to be taken in the event of radiological emergency, 27 November 1989.
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SUJB, May 2001 Assessment of Beyond Design and Severe Accident Consequences, Principles and methods of emergency planning and response at NPP Temelin, State Office for Nuclear safety, Praha, Czech Republic, May 2001, http://www.mzv.cz/EIA/eindex.html

- Abkommen zwischen der Regierung der Republik Österreich und der Regierung der Tschechoslowakischen Sozialistischen Republik zur Regelung von Fragen gemeinsamen Interesses im Zusammenhang mit der Nuklearen Sicherheit und dem Strahlenschutz, seit Juli 1990 in Kraft.
- Regulation No. 219/1997 Sb. of the State Office for Nuclear Safety on Details of Emergency Preparedness of Nuclear Facilities and Workplaces with Ionising Radiation Sources and on Requirements on the Content of On-site Emergency Plan and the Emergency Rules.
- Regulation No. 184/1997 Sb. of the State Office for Nuclear Safety on Radiation Protection Requirements.
- Decree of the Government of the Czech Republic No. 11/1999 on the Accident Planning Zone

10 Cluster "Safety Culture"

Issue 27: Safety Culture
Issue 28: NPP Organisational Structure and Management of Licensing

Table of Contents

10.1 Introduction

The broad issue of safety culture covers several aspects, which in general are difficult to assess. Having primarily operational safety in mind, two specific aspects were originally foreseen to be considered, including the way they are integrated into the management and operation of the plant.

(a) operating experience feedback and
(b) root cause analysis procedure

Agreements on how to proceed for the evaluation of these aspects were made dependent on the results of the OSART mission.

In addition elements have evolved from the activities in the frame of the trilateral discussions and related meetings which give concern about:

(c) the level of "safety culture" in the licensing process, which was in part envisaged to be covered under the Issue 28 ''NPP Organisational Structure and Management of Licensing''.

10.2 Identified Problems

Safety culture related to safe operation of NPP

After reviewing most of the safety relevant operational issues within the OSART process and in making use of additional information gained, concerns remain related to (a) operating experience feedback, and (b) root cause analysis procedure. No discussion of these aspects took place within the trilateral process.

The OSART Process [4] has revealed a number of areas in need of further improvement:

- Safety Culture development
- Self-assessment, quality assurance, performance audits, feedback, surveillance, maintenance, temporary modifications implementation

The OSART Mission treated Temelin Unit 1 as a unit in operation, even though the overwhelming portion of operational responsibilities still lay with the main contractor Skoda Praha and there was still extensive contractor work going on. The mission therefore evaluated the readiness for operation of the plant operator along with the start-up operations management provided by the main contractor. Therefore the Post-OSART Mission - scheduled approximately 18 month later - is considered equally important to gain a more comprehensive picture of the plant operator, in case transition to normal operation should have taken place then. The start-up workforce for Unit 2 has not yet been established.

Observations within the OSART Process were also made with regard to Safety Culture at the commissioning stage. There were numerous changes to technical specifications and temporary changes of considerable number and extent. Policies, development and procedures to control these safety relevant deviations in both numbers and duration were questioned.

In conclusion, to have a better insight into the aspects of safety culture related to NPP operation including NPP organisational structure and management, a specific meeting with the NPP is proposed.

The concern (c) level of “safety culture” in the licensing process relates to those items treated during the trilateral meetings and are connected with the licensing process, including related management, safety assessment, evaluation and documentation.

Safety culture related to the licensing process

The safety issues dealt with in the trilateral discussions under the Melk process were originally mainly concerned with matters of plant design and implementation. During the presentations and discussions, however, situations arose which raised concern regarding Operator's commitment and responsibility for nuclear safety, as well as the strength and independent evaluations capability of the Regulator. Thus the issue of safety culture related to the licensing process gained high priority. Some examples of observations causing concern were:

Role of SUJB and CEZ in trilateral meetings

During the meetings to discuss selected technical issues there was no instance in which SUJB showed that its role was distinct from that of the Operator, and CEZ in their presentation and discussions of the issues only rarely made explicit reference to the licensing process.

Roles and responsibilities

During the pertinent workshops situations arose showing that important documents (e.g., test protocols, non conformance reports) were not available at the plant, and that the operator was not in a position to answer questions e.g. on essential details of the ultrasonic testing of the RPV. It was unclear who was responsible for the safety of the plant, in order to answer the questions, at that specific time.

Licensing management by SUJB

In the above mentioned technical discussions, only few individual experts from SUJB took part, never the Project Manager for the licensing of Temelin NPP. During the discussions of open issues or pending aspects there was no evidence of effective control by the Regulator (evaluation performed, requests made for additional assessment, established action plan to resolve the issues and ways to keep them under control in the meantime, etc.).

Reference Regulations and Codes

Several times there was uncertainty among the Czech experts as to which normative basis was applicable, and how binding its character is (Czech Code (1998) “Guidelines and recommendations…”). In some instances, elements from different Codes (e.g. Czech, Russian, French, US) were merged and at least in one instance (seismic component qualification) there was no evidence of analysis made to determine the conservativeness of this composite approach.

Licensing requirements and documentation

There was evidence that important analyses and documents related to NDT (volume and quality of NDT in the primary circuit do not meet European practice and some non-allowable indications were discovered) and PTS (pre-service analysis fir Unit 1 has not been done), normally to be prepared before commissioning, were not performed or available, and there was no evidence of the SUJB position on these aspects before giving their authorisation for commissioning.

Non Conformity management

In the case of an indication discovered by NDT that was not allowable according to the Russian Code regulations for manufacture, and had about double the size of the Czech limiting value, use was made of the ASME Code. This has the appearance of evasion of Code requirements, which would have required repair before commissioning. In deviation from European- and internationally accepted practice, Equipment Qualification was not fully established before fuel loading. No information was given about the justifications brought forward by the Operator to receive authorisation for fuel loading without having equipment qualification fully established. Nor was there an indication of the basis and conditions for the Regulator to give authorisation to start nuclear commissioning. It was also not clear how the Operator is managing the ongoing programme for Equipment Qualification and endorsing his responsibility for demonstrating this qualification in front of the Regulator.

Regulator involvement and attitude

SUJB has been shown to have given almost no consideration to PSA study performed by CEZ for Temelin. Although PSA is not a licensing requirement and the Temelin PSA was not requested by the Regulator, there is no justification for the Regulator not making use of its findings for his own work (Compare Issue 6), provided that a suitable peer review is performed to validate the reasonableness of the PSA methods, assumptions, and results.[1]

The SUJB representatives were also passive during the discussions of key severe accident phenomena, which should be of primary concern to the Regulator in the definition of emergency planning.

Czech regulations[2] ask for consideration of all radiation accidents with a frequency higher than 10^{-7}/ year in the emergency planning (EP) on a national level. Without reviewing the PSA – how can SUJB be certain that all relevant accidents as required by the regulations were evaluated?

Although the safety problems at the 28.8 m level have been recognised a real concern, there was no satisfactory presentation of the problem in its overall importance, of the way it has been analysed, on the requirements established in this respect by the Regulator, on the results of investigations made and how the envisaged measures fit with the requirements to be satisfied and the physical layout.

Summing up, there is concern regarding the

- not fully satisfactory role of SUJB as a vital regulatory counterpart of Temelin NPP in addressing the extent and completeness of assessments, requirements and actions to resolve pending safety issues, and also through the effective management of the licensing process (addressing and integration of safety evaluations, internal as well as external support).

- not fully satisfactory role of the Operator in interpreting its unique responsibility for plant safety, the related licensing process and necessary safety justification, including effective coordination of external contributions (designers, suppliers, engineering organisations) and more dedicated documentation management.

- lack of discernible willingness on the part of the official representative of the Regulator to openly evaluate any concerns or suggestions put forward in the trialogue process.

[1] See H. Denton in Section 10.6 Attachment 1, Subsection 5. Recommendations
[2] Decree of the Government of the Czech Republic No. 11/1999 on the Accident Planning Zone

10.3 Solution to the Identified Problem

To discuss and resolve the Austrian concern the following is proposed:

- Have a complete understanding and evidence of main aspects of licensing of TNPP, the basis for authorisation to begin commissioning, and the manner in which the roles and responsibilities of Operator and Regulator have been implemented.
- Have the opportunity to have a more direct insight in the licensing management from the Regulator side and the Operator side.
- Have a specific meeting with TNPP about aspects related to safety culture of the operation and NPP organisational structure
- Support any initiative to define and finance from EC a Phare TSO project in support to SUJB (with their agreement) for Licensing of Unit 2
- Ways to strengthen the Regulatory Body should be found.

10.4 Observed Aspects and Indications

10.4.1 Introductory remarks

Although Safety Culture is well defined and is accepted to be fundamental to nuclear safety, its assessment is based on observations and indications from all activities related to ensuring plant safety.

Safety Culture is not only a requirement for the Utility but also for the Regulatory Authority. It is attitudinal as well as structural, relates both to organisations and individuals, and concerns the requirement to match all safety aspects and related processes (design, construction, licensing, operation, etc) with appropriate coherence, perception and actions. Its assessment is therefore a complex process, which cannot be limited to the study of documents or a walk down of a plant.

Meetings and discussion with SUJB and Temelin NPP devoted to licensing, management and safety culture (which permeates all activities related to design, construction, operation, licensing, surveillance and regulation of a nuclear installation) could have led to a more definite understanding of the status of this basic aspect of nuclear safety, but this was unfortunately not possible within the trialogue process.

It would be helpful to have this opportunity in the near future in order to get a direct insight into the way the roles and related responsibilities of Czech Nuclear Safety Authority and of the Temelin NPP Operator are defined and performed. These are key basic elements necessary to ensure the safe operation of Temelin NPP and effective assumption of responsibility for NPP safety by the Operator and effective regulation and surveillance by the responsible Czech Nuclear Safety Authority.

From the activity performed in the frame of the trialogue and related meetings for issue presentation and discussion, elements have been noted which give concern about the level of ''safety culture'' in the licensing process.

In particular these elements are connected with the licensing process and related safety assessment, evaluation and documentation. However, they were identified during the technical discussions of the limited number of items selected for these meetings - thus they do not cover all the aspects of the licensing process.

10.4.2 State-of-the-art requirements in Safety Culture

The term “safety culture” was first introduced by the International Nuclear Safety Advisory Group (INSAG) report on the Chernobyl accident in 1986. The notion of safety culture was subsequently expanded on in 1988 in INSAG-3, *Basic Safety Principles for Nuclear Power Plants*, issued in 1988. INSAG 3 advocates amongst others the following principle in paragraph 28: *'An established safety culture governs the actions and interactions of all individuals and organisations engaged in activities related to nuclear power.'* A multitude of consequences of this principle is formulated in many secondary guidelines.

In 1991, after increasing usage of the term, INSAG issued an entire report on safety culture as INSAG-4, which describes in detail aspects of safety culture with particular emphasis for the organisations involved in all activities connected to design, construction, operation, licensing and surveillance of nuclear installations.

INSAG defined safety culture as “*that assembly of characteristics and attitudes in organisations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance*”.

INSAG elaborated further to avoid misperception: *Good practices in themselves, while an essential component of Safety Culture, are not sufficient if applied mechanically. There is a requirement to go beyond the strict implementation of good practices so that all duties important to safety are carried out correctly, with alertness, due thought and full knowledge, sound judgement and proper sense of accountability.*

In particular, the Regulatory Authority is required to have adequate legal status, statutory authority, financial provisions, organisation, and attitude towards active pursuance and achievements in carrying out its regulatory duties; to have positive attitude to promote plant safety and protection of workers, public and environment; to have adequate staff and competencies in order to be able to make independent decisions and possibility to rely on external high level technical support independent from operator, if necessary.

The Regulator should at all times be transparent and able to demonstrate its independence in its regulatory activities and in any safety related decision that it makes.

10.4.3 Roles of SUJB and CEZ in trilateral meetings

During the meetings held in Vienna and in Prague to discuss selected technical issues there was no moment in which SUJB showed its role of Regulator distinct from the Operator. CEZ in their presentation and discussions of the issues only rarely made explicit reference to the licensing process and never reported facts related to this process which could have underlined the way the safety issue has been assessed according to SUJB requests and how additional or complementary analysis have been performed to match Regulator requirements.

It would have been appropriate for a Regulator having full consciousness of their independent function to find the way to clarify how they have been performing their role during the licensing (their requests, their enforcements, their conditions and so on). But SUJB always stayed in the background or confirmed NPP speakers (or their support organisation). Maybe they had elements to show in this respect, but they did not consider this significant for the discussion, even if the discussion was dealing with their own domain: safety aspects, safety requirements, safety analysis and so on.

This is to say that SUJB, during the discussion of these deficiencies and pending safety aspects, did not give the impression that they had complete control of the safety aspects related to the items under discussion, that they had required adequate safety justifications and the preparation of appropriate documentation by the Operator in order to proceed with the commissioning, and that they had established an agreed action plan to resolve the issues and keep them under control in the meantime.

The official SUJB representative in the meetings played a role much more bureaucratic than technical and always stood back, behind the operator or his contractors. The impression is that SUJB is not playing a sound effective role as the vital counterpart of Temelin NPP in addressing the extent and completeness of assessments, requirements and actions to resolve pending safety issues. SUJB appears just to have a rather bureaucratic position in the licensing process.

In some instances it was also evident that CEZ did not have complete knowledge and control of the safety basis, applicable regulations or limitations and extent of performed safety assessments. For many issues the presentation from CEZ have been made by representatives of other support organisations like REZ, VUJE, S&A. This does not mean that it is unusual for the Operator to rely on external support, but what appeared weak was the overall coverage and integration by CEZ of the safety aspects and related licensing (normative references, applied regulations, performed assessments, licensing process results and conditions, availability of design documentation and safety assessments documentation, etc.) i.e. the lack of endorsement of its unique responsibility for plant safety and its demonstration vis-à-vis the Nuclear Regulatory Authority.

10.4.4 Roles and responsibilities

During the workshops concerning Issue 09 (RPV embrittlement) and Issue 22 (non-destructive testing) situations arose in which it was unclear who was responsible for the safety of the plant, to answer the questions, at that specific time. Several questions could not be answered, neither by the utility nor by SUJB. Especially questions concerning essential details on the ultrasonic testing of the reactor pressure vessel (RPV) had to be passed on to Skoda Prague or Skoda Plzen.

Important documents (test plots with parameters of the ultrasonic testing and the test results) could not be reviewed because these documents are stored at Skoda's facilities. Also the question concerning the final reports on the mechanised ultrasonic testing of the RPV and the primary coolant circuit piping were insufficiently answered by the statement that Skoda is responsible for the completion of these reports. The Czech experts assumed that these documents would be transferred to the utility when commercial operation begins.

The situation was similar in the case of other important documents, i.e. the measurement protocols concerning the ductile-brittle transition temperature determination for the main circumferential welds of the RPV; only after repeated requests by the Austrian experts were these documents procured from Skoda and made available for inspection.

Another example is the non-allowable indication found in the Pressuriser (manufactured by Vitkovice): the corresponding non-conformance document is considered to be an indispensable part of the component documentation. But this document also had to be procured from the manufacturer to be made available for review.

From these facts the following questions arise:

- Who applied for the license for trial operation?

- How is CEZ managing the licensing process and related safety justifications and how is it co-ordinating the contribution of different designers and suppliers?

- Who holds the permit for the present phase of operation?

- How will this change with the onset of commercial operation?

- How is this organised at Temelin NPP during the in progress commissioning phase and how it will change passing to commercial operation?

- How is the plant documentation (in particular the documentation relevant for plant safety assessment and for each component important to safety including test reports, inspection reports and qualification reports) managed? What is not available onsite and why?

- Who is responsible for the complete documentation and the safety of the plant at present?

The lack of a clear distinction between the roles of the Licensing Authority, the Operating Organisation, the Plant Management, the Technical Support Organisations, the manufacturers and suppliers can have consequences detrimental on the ensurance of nuclear and radiological safety.

10.4.5 Temelin NPP Licensing management by SUJB

Not many experts from SUJB participated in the technical discussions of the selected issues. The SUJB licensing Project Manager for Temelin NPP was not introduced. According to earlier information, there was a Project Manager for licensing of Temelin NPP up to the beginning of commissioning Temelin NPP and then, it seems, this position was cancelled (or reduced in importance) as it was no longer considered significant for SUJB activity. This leaves the duty to follow the commissioning activity and results mainly to the site inspectors.

If this is confirmed (and it seems to be) this is very strange and far from an effective and efficient practice for the Regulator - especially considering that this is the first unit in Temelín NPP, and the plant is a prototype because of many design changes.

It is during the commissioning that the results of tests confirm the correctness of the assumptions, hypotheses, analyses performed in the POSAR [3] and the assessed behaviour of NPP in different plant conditions. The Project Manager for licensing from the side of the Regulator coordinated and addressed the safety evaluation during the licensing process and approved the POSAR chapters. He is the person best suited to integrate these results during the commissioning phase and their significance in the overall safety aspects of the NPP. He is of course an almost irreplaceable reference unless the integration of results is considered of lesser importance and surveillance is limited to single fragmented aspects (single tests).

10.4.6 Reference Regulations and Codes

Several times during the meetings and workshops it became evident that there was uncertainty among the Czech experts as to which normative basis was applicable.

Regarding the issues 09 and 22 there were explicit differences of opinion between the Czech experts about which Code should be applied for the evaluation of non-allowable indications observed during the pre-service inspection (NDT after the hot hydrotest).

According to experts from Rez the non-destructive testing (NDT) after the hot hydrotest is part of the start-up procedure and therefore the Czech regulations (“Guidelines for the evaluation of the lifetime of RPV and reactor internals of type WWER reactors during the operation of the NPP” [2]) have to be applied, while the representative of SUJB was convinced that exclusively the Russian Code regulations PK1514-72 have to be applied.

It is also important to note that the status of the Czech Code from 1998 is still unclear, three years after their issuance and well into the commissioning procedure of Temelin NPP. It is still called a draft but SUJB recommends its application. The introduction to this document states:

“The ‘Guidelines and recommendations....’ are meant to be guidelines for the licensee for the preparation of a safety documentation according to the attachment of the law No. 18/1997 Slg., part E and F, for the lifetime assessment (integrity evaluation) of the reactor pressure vessel and the reactor internals. The draft of the ‘Guidelines and recommendations....’ will be the basis for the final version, that will be edited by SÚJB as ‘Safety Guidelines’ on the basis of objections and comments. If the licensee will be using the final version of ‘Guidelines and recommendations for the evaluation of the reactor pressure vessel and the reactor internals of WWER NPP during the operation of the NPP’ after it’s edition the respective part of the safety documentation will be accepted as adequately fulfilling of the legal requirements.”

The situation is even more confusing in the case of the seismic and component qualification. In the presentation of the Czech experts 8 normative Codes were named for the seismic qualification, 5 normative regulations for the component qualification. A question concerning the normative basis of the seismic component qualification started an investigative effort on the Czech side and was later answered in a written format. According to this the demonstration of the seismic strength is performed with the following methodology:

The complete component and piping system is calculated as piping statics using the calculation code SYSPIPE Cod, version 231c from FRAMATOM_FRAMSOFT/CSI, certified for the Czech Republic by SUJB. Based on these calculations, the stresses within the components are calculated according to the Russian Code G [1]. For a T-type pipe this can be performed using two different methods:

- a simplified calculation procedure: attachment 5 in [1]: calculation of typified components and devices (this attachment is of recommendatory character), paragraph 2.3: piping systems with low temperature loads

- in depth calculation method: attachment 5 in [1]: calculation of typified components and devices (this attachment is recommendatory character), paragraph 2.9: calculation of stresses in a T-type branch using more precise methodology

In a third step FEM calculations of the component can be used for an even more precise calculation of the stress distribution. The resulting stress categories are compared with the respective limiting values in the ASME-code, Section III, NB.

This means that a combination of three different Code regulations with completely different origin and development history is applied. This requires a critical analysis, e.g. a comparison of the initial assumptions (selection of the design earthquake, validation of the response spectra, load assumptions) within the different Codes. This is necessary because the individual national normative Codes always assume a relation between the different steps. For instance, in case of very precise or very conservative initial parameters the required safety margins can be rather low, while these would lead to completely wrong conclusions in case of simplified or non-conservative initial parameters. Especially the selection of the maximum credible earthquake according to different assumptions in the Codes and the requirement of precision in the calculation of response spectra strongly influence the required safety factors for the definition of the allowable stresses.

A comparative analysis especially with respect to conservatism of the methodology used in the commissioning of Temelin NPP has not been performed up to now.

10.4.7 Licensing requirements and documentation

The following severe deficiencies were found with respect to the preparation of the documentation as part of the licensing procedure:

- Important reports on the pre-service non-destructive testing (final reports on the mechanised ultrasonic testing of the RPV and the primary coolant piping) were not available. There was no summarising document on the pre-service NDT of the safety relevant components of the primary and secondary circuit. Such a document must including a comparison of the legally required testing volume and the actually performed testing extent, a description of the testing methodologies, especially their sensitivities, and a list of all indications above the registration limit. Only sporadically (isolated) protocols with strongly reduced information were found as part of the component passports.

- Some of these were not available at the utility, so no complete overview of the quality of manufacture and assembly and the integrity of the safety relevant components could be achieved.

During the review of several documents three non-allowable indications were found by the Austrian experts that were not named in the summarising presentations of the Czech experts. A complete follow-up on these indications was not possible due to the non-availability of the documentation in the utility. But there is almost no doubt that these indications were left unrepaired without the performance of additional safety analyses as is required by the Czech “Guidelines and recommendations....”.

Although technology and know-how were available in-depth non-destructive testing was not performed for these indications. The non-allowable indication in the RPV was classified as below the registration limit after additional manual ultrasonic testing. This kind of approach can easily lead to unacceptable “playing with the amplitude” and is in contradiction to European safety culture. It is questionalble whether the legal requirements of the Czech Republic are met by this procedure.

The question arises of how these facts were evaluated by SUJB and what has been the Regulators position? There was no indication of either in the documents as made available.

Another important requirement for commissioning is completely missing: the structural integrity assessment of the RPV for pressurised thermal shock conditions (PTS analysis) under consideration of the neutron embrittlement of the RPV steel. This analysis is internationally required as part of the licensing documents, also required according to the Russian Code and the Czech “Guidelines and recommendations...”. In addition the IAEA issued recommendations combining the knowledge on the Russian materials and the Russian project with modern European safety standards and calculations methods, that also require a PTS analysis before start-up. Such a pre-service PTS analysis was not performed for Temelin NPP; according to the Czech side it will be performed for both units within the next 5 years.

Again, there was no indication – neither in the presentations nor in the documents made available - of the position taken by the Regulator or the justification for commissioning to the present stage of trial operations without the fulfilment of the above requirement.

Summarising, there was evidence that important analyses and documents related to NDT as well as PTS analysis normally to be prepared before commissioning were not performed or available and there was no evidence of how these aspects were considered by SUJB in giving their authorisation for commissioning.

10.4.8 Non conformity management

A case that has a strong appearance of evasion of Code requirements was observed in the evaluation of a non-allowable indication in the lower head of the RPV. This indication was neither allowable according to the Russian Code regulations for manufacture, nor according to the Czech Code regulations said to be valid for start-up procedure (compare 10.5.6). The indication had about double the size of the Czech limiting value. By making use of the ASME Code that allows such defects this difficulty was by-passed and a repair (which would have delayed the commissioning) was avoided.

It was again completely ignored that national Codes always need to be an integrative unit reflecting national traditions and realities. For the WWER-1000 e.g. materials and manufacturing methods, strength and safety design differ strongly from US practices. Therefore “norm picking” (selection of the more convenient Code regulation) - as applied in the above example - must be rejected on principle. In the specific case, the procedure was obviously accepted by SUJB, thus bypassing their own regulations in “Guidelines and recommendations....” without documentation of an appropriate justification for this decision.

In Temelin NPP EQ (equipment environmental and seismic qualification) was not fully established before fuel loading and still is not completed. A complex programme for environmental re-qualification of safety equipment and components has been defined and is to be implemented between 4/2000 and 6/2002.

According to European practice and requirements the qualification of equipment important for safety has to be ensured and demonstrated before the installation of equipment and in any case the formal qualification reports should be issued and approved before nuclear tests.

For equipment important to safety (safety systems and safety related systems) the environmental and seismic qualification is evidently delayed to a date well beyond the plant start-up. The NPP’s test operation with equipment of credible instead of sufficiently demonstrated qualification must be considered a specific deviation requiring a timeframe for the compliance to be established, demonstrated to and accepted by the licensing authority.

Information about the current status of this programme was presented during the meetings by Stevenson and Associates. But no information was given on the technical basis by which on the one side the utility applied for commissioning in spite of this issue being open and by which SUJB on the other side authorised this.

In addition the presentation in Prague mainly gave information about the qualification of safety systems. It is reported that a decision from SUJB gives priority to the process of environmental qualification for safety systems and the same process for safety related systems will be continued subsequently.

Information about the current situation of qualification of safety related systems and the planning of the process for full establishment of their environmental qualification was not clarified in the presentation.

Complementary information needs to be acquired on this aspect together with a review of qualification reports on site.

The POSAR [3] states on page 3.11.3 that *‘’a Contractor will be in charge of implementing the complex programme for qualification and he will endorse the responsibility to defend the results in front of Czech Regulator’’.* This raised doubts that the Operator is taking a primary role in this activity. The primary responsibility for qualification can never be delegated to a contractor: a contractor can perform tests and analyses but the responsibility to ensure the qualification of equipment operating in the plant belongs to the Operator. This means that the Operator has to demonstrate his own culture and knowledge in the matter of qualification, manage all the aspects related to equipment qualification, integrate the work performed by different contractors ensuring correct interfaces and demonstrate the correct implementation of this safety requirement to the Regulator.

The presentation of this issue from Czech side was not made by NPP, but by a contractor. There was no indication also of the position of SUJB with respect to the issue of equipment qualification and related past and in progress licensing process.

10.4.9 Involvement of the licensing authority

10.4.9.1 PSA and Emergency Planning

Probabilistic Safety Assessment (PSA) is an important part of the safety assessment of nuclear power plants and an important tool to

- support decision,
- have insight into the plant defences balance with respect to different events and initiators,
- have an insight into the plant and system weaknesses
- provide extensive information on initiators, propagation and vulnerabilities to SA
- provide support in identifying plant updates having potential to significantly reduce residual risks, etc.

PSA is not a licensing requirement according to Czech regulations and accordingly the Temelin PSA was not requested by the Regulator. It was performed for Temelin NPP on the initiative of the Operator.

SUJB stressed that the plant has been licensed on the basis of deterministic safety standards and traditional design basis analyses - which is an accepted approach, common also to other countries. Even so, there is no justification for the Regulator not to make use of the PSA findings for his own work (Compare Issue 6).[3].

In fact, PSA is required by US regulations and is frequently requested by Regulators in European practice, because probabilistic analyses can give important overview and insight in complex systems such as nuclear power plants.

There was evidence that SUJB had no active role in the definition of requirements of PSA and in practice they have performed no review of it (it was only mentioned that one expert from SUJB took part in one IAEA review mission). The involvement in the PSA study from SUJB has been minimal.

As SUJB seems to be almost ignoring PSA, it is understandable why this study has not yet been reviewed by the Operator since 1995. At that time many preliminary assumptions were made which now should be reviewed (more specific models, data, etc.) in order to have a more coherent, updated and useful PSA.

The SUJB representatives were also passive during the discussions of key severe accident phenomena such as probabilities, containment failure modes, and the associated uncertainties, - PSA results and spin-off – which, nevertheless, should be of primary concern to the Regulator in the licensing process and in emergency planning.

[3] Compare Section 10.7 Addendum 1, Subsection 5. Recommendations

This gives the concern regarding the suitability of SUJB's attitude and its willingness to:

- promote effective use of this study in the decision process (to identify weaknesses, to make comparison, to define priorities, to define importance and so on)

- exercise regulatory oversight over the Temelin PSA development

- emphasise the value of PSA insights within its own organisation.

Active SUJB involvement in the PSA process is especially essential since there are significant design differences between Temelin NPP, a Soviet designed WWER supplied by Skoda with important changes provided by western designers, and the PWR’s of OECD member countries.

Oversight by a regulatory body whose primary purpose is safety can stimulate complete, objective analyses and comprehensive improvements. The lack of active SUJB involvement in the PSA process is a significant regulatory shortcoming.

In addition, SUJB has a key role in emergency planning on the national level (center of radiation monitoring, the crisis coordination center, international data exchange and determination of the Emergency Planning Zone ‘’EPZ’’). The requirements for determining the EPZ are laid down in Governmental Decision No. 11/1999. According to this Decision all radiation accidents with a frequency higher 10^{-7}/year have to be taken into account for determining the EPZ (probabilistic perspective). In other words the source term to be assumed for the definition of the EPZ is that coming from sequences having a frequency equal or higher than 10^{-7}/year.

Thus PSA is the only available basis for the identification of sequences of interest. As the PSA was not reviewed by SUJB, there is concern about the appropriate importance given by SUJB to PSA also for the EPZ requirement (defined by Governmental Decision) and about the way they have evaluated the correctness and completeness of the severe accident analysis and related methodologies, models and data.

10.4.9.2 High Energy Line Break Issue

At level +28.8 m a real concern exists with respect to physical separation among steam lines and feedwater lines, which is not in compliance with western approach and requirements. The rupture of a steam line could lead to damage to other lines or components not physically separated.

Limited measures taken to reduce this risk (instalment of restraints) were presented, but there was no satisfactory presentation of the problem in its overall importance, of the way it has been analysed, of the requirements established by the Regulator, of the results of in-depth investigations made on how the envisaged measures fit with the requirements to be satisfied and with the physical layout.

SUJB again stayed in the background; their position was not made clear.

10.5 References

[1] PNAEG-7-002-86: Russian Code, 1986.

[2] Guidelines and recommendations for RPV and reactor internals lifetime assessment (SUJB), Draft (December 1998)

[3] Temelín POSAR, Rev. 1, December 1999.

[4] Report of the OSART Mission to the Temelin Nuclear Power Plant 12 February - 1 March 2001, IAEA, May 2001, Vienna

- INSAG 3 –1988: International Nuclear Safety Advisory Group, “Basic Safety Principles for Nuclear Power Plants”, INSAG-3 Report, IAEA Safety Series No. 75, 1988.

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- INSAG 8 -1995: International Nuclear Safety Advisory Group, “A Common Basis for Judging the Safety of Nuclear Power Plants Built to Earlier Standards”, INSAG-8 Report, IAEA Safety Series No. 75, 1995.

- INSAG 13 - 1999: International Nuclear Safety Advisory Group, “Management of Operational Safety in Nuclear Power Plants”, INSAG-13 Report, IAEA Safety Series No. 75, 1999.

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- IAEA-WWER-SC-171: IAEA, Review of WWER-1000 Issues Resolution at Temelín, WWER-SC-171, IAEA-TA-2490, TC Project RER/9/035, Temelin, Czech Republic, 11 to 15 March 1996

10.6 Attachment 1: Mitigating Temelín Vulnerability to Severe Accidents

An assessment by Harold Denton

1. Introduction

On April 3 through 6, 2001, I participated in meetings in Vienna and Prague regarding the Temelín Trialogue process and Environmental Impact Assessment process on behalf of The Institute of Risk Research. This report provides my evaluation and recommendations regarding the safety assessments being performed in the Czech Republic to evaluate and reduce the risks from severe reactor accidents at Temelín Units 1 and 2.

2. Approach To Severe Accidents In OECD Member Countries

Since the reactor accidents at Three Mile Island and Chernobyl, detailed consideration of severe accident vulnerabilities has become a standard practice in Europe and the United States.

The extensive information that has been generated internationally regarding the initiation and propagation of severe accidents has permitted the characterisation of residual risks and the identification of plant enhancements that have a potential for significantly reducing such risks. Of particular importance are the development of a set of accident frequencies and their occurrence frequencies, the mode and timing of containment failure relative to reactor vessel failure, the projected consequences and the emergency response measures.

For example, the U. S. Nuclear Regulatory Commission announced in 1985 a plan to address severe accident issues for operating nuclear power reactors in the United States. Specifically, all licensees were requested to identify vulnerabilities to severe accidents and consider implementation of cost effective mitigation measures for instances of particular vulnerability to core damage or poor containment performance. In response, the NRC received 75 Probabilistic Safety Analyses(PSA) covering 108 plants. Licensees were not required to calculate offsite health effects although a few did. A total of over 500 proposed improvements were identified, with specific improvements varying from plant to plant. In the Final Environment Statement for the operation of the Watts Bar Nuclear Plant in 1995, for example, 26 candidate design improvements were assessed and three were selected for implementation.

Also, the Nuclear Energy Agency has been active in developing and co-ordinating severe accident activities within OECD member countries since about the mid 1980’s. NEA announced in 1996 the continued commitment within OECD countries toward prevention and mitigation of severe accidents in nuclear power reactors and reported that plant modifications and changes were and continued to be implemented to reduce risks from severe accidents. A report in 1997 compiled a general listing of examples of “software” and hardware” plant modifications, which had been implemented in Finland, France, Germany Netherlands, Spain, and Sweden, Switzerland, United Kingdom and other OECD member countries. Level 1 and 2 PSAs, and in some cases Level 3, were used to identify the dominant contributions and most significant modifications. Although many OECD countries consider plant specific assessments in their decision process, formal regulatory guidelines and procedures are not in widespread use.

3. Temelin Approach to Severe Accidents

On April 4, I attended a meeting in Prague on the technical basis for Temelin emergency planning zones. Presentations by Czech staff from SUJB, VUJE and the Temelin NPP included the technical and legislative bases for emergency planning, design basis events, severe accidents, radiological consequences and emergency preparedness.

At the beginning of the meeting, Mr. Krs, SUJB, stressed that the plant has been licensed on the basis of deterministic safety standards and traditional design basis analyses, and had met all applicable Czech statues and regulations. Mr. Holan of the Temelin NPP took a similar position although the utility had completed a Level 1 and 2 PSA as early as 1996 and apparently was continuing to update and revise the results. Neither individual exhibited an attitude likely to bring severe accident management to closure for Temelin in the near future. For example, near the end of the meeting, in response to questions, the VUJE representatives indicated that cost effective modifications to mitigate severe accidents could be available in about 18 months. Messieurs Krs and Holan both took exception to that estimate, citing higher priority work.

Mr. Prouza, VUJE, discussed Temelin’s event classification scheme, emergency planning zones and preplanned actions, the national radiation monitoring network and the emergency response organisation. It appeared to me that planning was detailed and appropriate, and not that different than that in many OECD member countries. Apparently the Foreign Office is responsible for contacting organisations outside the Czech Republic such as the IAEA and the OECD in event of an emergency. It was not clear to me whether Austria would be contacted directly by the Foreign Office or not.

Messieurs Bujan and Stubna, VUJE, discussed the methodology and results of various severe accident scenarios. Two “worse case” analyses were presented, the AB and the V sequences. The AB sequence assumed a large LOCA, a total loss of offsite and onsite electric power, and no operator intervention. This led to an essentially adiabatic heat up of the core, fuel melting beginning at 31 minutes, reactor vessel bottom head failure at 104 minutes, and containment vessel failure about 18 hours later. All noble gases were assumed to leak out within four days. In the V sequence, a break of the steam generator hot collector cover header was assumed, resulting in containment bypass leakage. Once again a station blackout was assumed, and no credit was taken for operator intervention. In this case, the vessel failed at in 90 minutes. The most important leakage of fission products occurred during the in vessel melting phase, from 35 minutes until vessel failure.

In my opinion, many cost effective improvements are likely to be found practical in Temelin that could reduce the probabilities of major releases for dominant accident scenarios. For example, if core damage from the AB sequence could be arrested in-vessel using accident management procedures such as are now in place at TMI Unit 1, the probability of containment failure becomes very small (Accident management involves the development of procedures and guidance to assure the most effective use of all available plant equipment and staff in the event of an accident). Containment basemat melt-through may be significantly delayed by adopting, for example, the Sizewell B approach of adding water to the reactor cavity using the containment spray system, the containment fire suppression system or by gravity drain from the refuelling water storage tank. A means of spreading core debris could also significantly prolong the maintenance of containment integrity. Special operator training for using primary side feed/bleed techniques in coping with the V sequence may direct most fission products from the core into the containment. The Comanche Peak PWR in the U.S. identified and evaluated over ten potential design alternatives for mitigating severe accidents, including reactor coolant system depressurisation, additional instrumentation for bypass sequences, and an independent containment spray system, as well as a cavity flooding system.

After the meeting on April 4, I learned that the original PSA conducted by the Temelin had been carried out using an early NRC code. I understand that there have been some modifications made by the plant on the basis of these results to reduce core damage frequencies but none involving the containment. With the involvement of Westinghouse in the plant, the latest severe accident computer codes used in OECD member countries have become available to the plant, including MELCOR and RELAP 5 Mod 3.2, developed for the NRC and ATHLET developed by the GRS. A full update of the level 2 PSA using these codes was started recently and is expected to be finished by mid 2002.

4. Role of SUJB

While the efforts by the plant to understand severe accident vulnerabilities has been long standing and positive, the involvement of the Czech regulatory body appears to have been minimal. The SUJB representatives in the April 4 were passive during the discussions of key severe accident phenomena such as probabilities, containment failure modes, and the associated uncertainties. Apparently, SUJB feels constrained to not extend their regulatory authority beyond that of a traditional “design basis” approach. This is inconsistent with the practice of regulatory bodies in many OECD member countries of considering information from plant specific PSAs regularly in their decision process. Although PSA is generally not the only basis for deciding whether or not to perform modifications it is sometimes the dominant one.

Active SUJB involvement in the process is essential since there are significant design differences between Temelin, a Soviet designed WWER supplied by Skoda, and the PWR’s of OECD member countries. The Temelin PSA models at this stage are less developed than for the PWRs of OECD member countries, lack the multiple international comparisons, and are thus more uncertain. Oversight by a regulatory body whose primary purpose is safety can stimulate complete, objective analyses and comprehensive improvements. In my opinion, the lack of active SUJB involvement in the PSA process is a significant regulatory shortcoming.

5. Recommendations

A. That an updated Level 1 and 2 PSA for Temelin reflecting consideration of current state-of-the-art approaches and issues for PWRs with large dry containments be completed that would include explicit identification of specific vulnerabilities to severe accidents that could be mitigated with cost effective improvements.

B. That an independent peer review of the scope, methods and rigor of the resulting PSA for Temelin be conducted by SUJB or under their auspices, focusing especially on containment vulnerabilities.

C. That identified “high” risk reduction enhancements in operational and emergency procedures, operator training and plant design be implemented to the extent practical at the first refuelling of Unit 1 and prior to the startup of Unit 2.

D. That the insights from the PSA be used to establish organisational and procedural measures for coping with severe accidents from an operator’s perspective.

E. That SUJB actively follow Temelin’s development of an updated Level 1 and 2 PSA, and monitor implementation of resulting mitigating actions. Further, SUJB should evaluate the results of the overall severe accident programme at the end of the process and document its own safety conclusions and recommendations.

6. Conclusions

It is my opinion that the Czech Republic has completed only in part the severe accident effort needed to demonstrate whether severe accident risks from Temelin are comparable to those in OECD member countries. In particular, the vulnerabilities to severe accidents that could be fixed with cost effective improvements have not been explicitly identified, evaluated or implemented. Also, the SUJB has neither exercised regulatory oversight over the Temelin PSA development nor emphasised the value of PSA insights within its own organisation, as have OECD member countries. However, with completion of the above recommendations, and with a resultant showing of a low core damage frequency and a low conditional probability of early containment failure, Temelin could be considered comparable to nuclear power reactors in OECD member countries with respect to the consideration of severe accidents.

Also, full Austrian participation with the Czech Republic in “large release” emergency exercises involving Temelin and Dukovany could be valuable. The feasibility of a using a direct nuclear data link with the Emergency Center of SUJB in such drills should be considered. Periodic drills would help assure that adequate communications and workable protocols exist with the Czech Republic, and allow testing of the Austrian national emergency programmes and decision making process for countermeasures. In my experience, the Austrian focus should be on obtaining accurate, timely and complete information about real or potential releases.

Harold R. Denton
Nuclear Safety Consultant April 23, 2001

ALARA (principle)	As Low As Reasonably Achievable (Principle)
ASCOT	Assessment of Safety Culture in Organisations Team
ASME	American Society of Mechanical Engineers
ATHLET (code)	Analysis of THermal-hydraulics of LEaks and Transients
ATWS	Anticipated Transients without Scram
BDBA	Beyond Design Basis Accident
BRU-A	Main Steam Relief Valve
CDF	Core Damage Frequency
CFR	Code of Federal Regulations (US)
CSNI	Committee on the Safety of Nuclear Installations (OECD)
DAC	Distance Amplitude Correction (curve)
DB	Double Ended Break
DBA	Design Basis Accident
DCH	Direct Containment Heating
DID	Defence in Depth
DBE	Design Basis Earthquake
DG	Diesel Generator
DGS (diagrams)	Distance Gain Size (diagrams)
EBP	Extrabudgetary Programme (IAEA)
EC	European Commission
ECCS	Emergency Core Cooling System
EdF-CUMULUS	Test facility of Elictricité de France (EdF)
EGP	Energoprojekt Prague
EIA	Environmental Impact Assessment
EMC	Electromagnetic Compatibility, Emergency Management Center
EOL	End-of-Life
EOP	Emergency Operating Procedure
EP	Emergency Planning
EPR	European Pressurized Water Reactor
EPRI	Electric Power Research Institute
EPZ	Emergency Planning Zone
EQ	Equipment Qualification
EURATOM	European Atomic Energy Community
EZ	Earthquake Load/Czech version
FEM (calculations)	Finite Element calculations
GILs	Generic Intervention Levels
GSHAP	Global Seismic Hazard Assessment Program
GRS	Gesellschaft für Reaktorsicherheit und Anlagen
HDR	Heissdampfreaktor
HELB	High Energy Line Break
HPME	High Pressure Melt Ejection
I	local (earthquake) intensity at a given distance from the epicenter
I_0	epicentral (earthquake) intensity
I_{0max}	maximum credible epicentral intensity
I_{max}	maximium credible local intensity
I&C	Instrumentation and Control

IAEA	International Atomic Energy Agency
ICISA	International Commission for Independent Safety Analysis
INSAG	International Nuclear Safety Advisory Group
IPERS	International Peer Review Service (IAEA)
IRR	Institute of Risk Research
ISI	In-Service Inspection
ISP	International Standard Problem
KTA	Kerntechnischer Ausschuß
LBB	Leak Before Break
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LPZ	Long Term Emergency Planning Zone
LWR	Light Water Reactor
MCE	maximum credible earthquake
M_{max}	maximum earthquake magnitude
MP	Melk Protocol
MPa	Megapascals
MSK	Medvedev-Sponheurer-Karnik (seismic intensity scale)
MSSV	Main Steam Line Safety Valves
MSIV	Main Steam Isolation Valve
ND*	low pressure sequence (late depressurization of primary loop by accident management)
NDT	Non-Destructive Testing
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute (US)
NPI	Nuclear Power International
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRI	Nuclear Research Institute
NSC	Convention of Nuclear Safety
NUPEC	Nuclear Power Engineering Corporation
OECD	Organisation for Economic Co-operation and Development
OSART	Operational Safety Review Teams
PAR	Passive Autocatalytic Recombiners
PAZ	Precautionary Action Zone
PGA	Peak Ground Acceleration
PHARE	Poland-Hungary Aid for Reconstruction of Economies
PISC	Programme for the Inspection of Steel Components
PORV	Power-Operated Relief Valve
POSAR	Pre-Operational Safety Analysis Report
PRISE	Primary to Secondary Leak
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PSHA	Probabilistic Seismic Hazard Assessment
PTS	Pressurised Thermal Shock
PWR	Pressurised Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking

QME	Qualification of Mechanical Equipment
QV	Qualification of Valves
RCC-M	French regulations
RDB	Reaktordruckbehälter
RELAP	Thermohydraulic code used for simulation of transient conditions in PWRs
RBMK	Reaktor Bolshoi Moshchhnoski Kanalkni (High-Power Channel Type Reactor)
RSK	Reaktor Sicherheits Kommission
RPV	Reactor Pressure Vessel
SA	Severe Accident
SAM	Severe Accident Management
SAMGs	Severe Accident Management Guidelines Richtlinien zum Management schwerer Unfälle
SC	Safety Culture
SCSIN	Service Central de Sûreté des Installations Nucléaires (Central Department for the Safety of Nuclear Installations)
SEOP	Symptom-oriented Emergency Operating Procedures
SG	Steam Generator
SGCL	Steam Generator Collector Leakage
SGTR	Steam Generator Tube Rupture
Siemens KWU	Siemens Kraftwerk Union AG
SKI	Statens Kårnkraftinspektion Swedish Nuclear Power Inspectorate
SOAR	State of the Art Report
SONS	State Office for Nuclear Safety
SORVs	Stuck Open Relief Valves
SSE	Safe Shut Down Earthquake
STCP	Source Term Code Package
STUK	Säteilyturvakeskus
SUJB	Státní úřad pro jadernou bezpečnost (State Office for Nuclear Safety)
TACIS	Technical Assistance to the Commonwealth of Independent States
TNPP	Temelín NPP
TOFD	Time Of Flight Diffraction
TRL Probe	Transmitter-Receiver Longitudinal Probe
TSC	Technical Support Centre
TÜV	Technischer Überwachungsverein
UBA	Umweltbundesamt
UPZ	Urgent Protective Action Zone
USAEC	U.S. Army Environmental Center
UVE	Umweltverträglichkeitserklärung
VUJE	Vyskumny ústav jadrovych elektrarni (Trnava, Czech Republic)
WENRA	Western European Nuclear Regulators' Association
WOG	Westinghouse Owners Group
WWER (VVER)	Water Water Energetic Reactors

ZAMG	Zentralanstalt für Meteorologie und Geodynamik
ZfP	zerstörungsfreie Werkstoffprüfung