umweltbundesamt

ETE Road Map

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

Item 3
Reactor Pressure Vessel Integrity and Pressurised Thermal Shock

Final Monitoring Report

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

Vienna, June 2005

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EXECUTIVE SUMMARY

I. Basis and the background for the project

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

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

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

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

The Austrian Federal Ministry of Agriculture, Forestry, Environment and Water Management entrusted the Umweltbundesamt (Federal Environment Agency) with the general management of the implementation of the "Roadmap". Each entry to the "Roadmap" corresponds to a specific technical project.

The Roadmap project treated here is focused on the exchange of information related to:

Item No.3: Reactor Pressure Vessel Integrity and Pressurised Thermal Shock (Workshop scheduled for the first half of the year 2004): *"This topical meeting will serve to address the status of the PTS (Pressurized Thermal Shock) analysis."*

The objective of the Roadmap process covered by this Roadmap Item as stated in ANNEX I of the "Brussels Agreement" is:

"The reactor pressure vessel (RPV) integrity under pressurized thermal shock (PTS) conditions shall be maintained with sufficient safety margin against brittle fracture throughout the NPPs service life."

In addition ANNEX I provides the following statements regarding the "present status and specific actions planned":

"The NPP Temelín is commissioned and operated respecting pressure-thermal (PT) curves calculations developed according to Westinghouse methodology. These calculations will be expanded with set of further PTS analysis for both units using a step-by-step approach with full respect of the IAEA Guidelines for the PTS analysis. The PTS analysis will be finished in accordance with approved project work plan for this item."

On behalf of the Austrian Government the Umweltbundesamt (Federal Environment Agency) committed an Experts' Team composed of international experts to provide technical support for the monitoring of the implementation on the technical level of the RPVI – PTS Issue as listed in ANNEX I of the "Conclusions of the Melk Process and Follow-up". This specific technical project is referred to as project PN9 comprising altogether seven predefined "project milestones" (PMs).

A Specialists' Workshop on the Roadmap Item No. 3 "Reactor Pressure Vessel Integrity and Pressurised Thermal Shock" was conducted in Prague on May 24 and 25, 2004 according to Article 7 (4) of the Bilateral Agreement of the Exchange of Information on Nuclear Safety. The workshop information was supplemented later, on October 7, 2004, by presentations given at Řež, which provided additional detailed information and answers to questions. These workshops were key elements in the monitoring process. The analysis by the experts and the team of information made available during the workshops played a significant role in the development of the Final Monitoring Report prepared by the Austrian Experts' Team.

The technical support for the monitoring on the technical level of the implementation of the "Conclusions of the Melk Process and Follow-up" regarding the item Reactor Pressure Vessel Integrity and Pressurised Thermal Shock Issues was aimed at focussing on the evaluation of how the Czech Side (operator and regulatory body) has dealt and will deal with the issue in a methodological way for implementation. In particular, it was intended to focus on the implementation of surveillance programmes and comprehensive PTS analysis, all to be checked against the background of requirements and practices widely applied within the EU and of new developments in WWER-reactor specific knowledge, both on the technical and regulatory level.

II. The approach and objectives of the PN9 project

The Temelín NPP, originally of Soviet design, and later upgraded to include elements of western safety concepts and western equipment, has addressed PTS and RPV integrity late in the construction phase. Russian and Western Codes request a pre-service PTSA. During the Experts' meetings in the frame of the Melk process it appeared that the process of PTS prevention implementation at Temelín was late and still not complete. The availability of information on the details of the approach adopted at the Temelín NPP was insufficient. Therefore, PTS remained one of the items to be addressed during the follow up to the Melk process. This established the basis and defined the scope of the proposed project.

The NPP Temelín has to be considered as a very specific case: Design and construction were performed in the former Soviet Union, the manufacture occurred at least partially in the former Czechoslovakia under Russian supervision. After the political re-organisation of Eastern Europe the construction was completed including Western technology from Westinghouse under the responsibility of the plant owner. Licensing happened within the legal frame of the Czech Republic.

The project PN9 „Reactor Pressure vessel (RPV) Integrity and Pressurised Thermal Shock (PTS)" deals with the topic of RPV damage especially as a consequence of a possible thermal shock transient. In the case of most critical transients, the primary circuit is under high pressure. This is one of the main concerns within the reactor safety analysis, since the RPV pressure retention and radioactive inventory retention functions are of non-redundant nature by design. A rupture of this component would therefore induce a catastrophic accident.

Consideration of RPV integrity (RPVI) as well as the exclusion of the PTS (pressurized thermal shock) at the Temelín NPP (nuclear power plant) is an essential ingredient of its defence in depth approach and therefore of utmost importance to Austria.

PTS events should have very low frequencies, since they may have significant consequences resulting in failure of at least one entire barrier (the primary coolant system envelope). As PTS has not been explicitly considered in the design of many older nuclear power plants, which are currently in operation, considerable efforts have been devoted already by most of those plants to prevent PTS events during plant operation, but also at zero-power, shutdown and during outages. PTS prevention has been recognised as an important safety issue at large and is consequently addressed in a comprehensive and systematic way.

In applying current safety philosophy, the consideration of PTS and RPV-integrity in NPPs usually includes the precautions taken to avoid excessive embrittlement, RPV material degradation and PTS sequences.

The project PN9 is composed of two complementary segments (horizontal and vertical), the horizontal segment depicting an assessment of principles, standards and practices, the vertical segment providing an analysis of PTS bounding cases.

In the light of the broad scope of PN9 not only the effort, but also the result addresses all the disciplines [ANNEX A] that were covered in the monitoring and at the Workshop as well.

The monitoring process conducted by the Experts' Team was concentrated on the engineering approach taken by CEZ to have the Temelín RPVs licensed by the SÚJB (State Office for Nuclear Safety).

Both segments are related to the collection of information on the Temelín RPV embrittlement behaviour over time, as well as the vessel's material history and usage and its thermal shock vulnerability.

III. Preparatory and Main tasks accomplished within the PN9 project

The main tasks to be accomplished by the Experts' Team were those in fulfilling the **Project Milestones (PMs)** ordered by the contracting party, the Umweltbundesamt. Several preparatory tasks had to be performed to support and accomplish the main tasks. These preparatory tasks are also addressed here.

For the different tasks within an RPVI assessment the state-of-the-art practice was reviewed for comparison with the findings of the evaluation of the Czech approach. RPVI includes the following steps:

- RPV quality with respect to design, construction/manufacture
- PTS analysis
- Surveillance programme
- NDT programme
- Core modifications
- EOPs

At the time of the NPP Temelín RPV construction the actual knowledge on radiation embrittlement was that copper and phosphorus impurities were causing the problematic irradiation embrittlement behaviour of the WWER RPV steels. Therefore for the WWER-1000 RPVs the steel was purified with respect to copper and phosphorus. In order to reach better hot work manufacturing properties the content of the alloying element nickel was increased. Only years later, it turned out that it might be this high nickel content that introduced a new kind of embrittlement mechanism. Modern RPV material – to be manufactured according to the state-of-the-art – would be steel optimised for minimum radiation embrittlement susceptibility.

It is therefore problematic to compare the Czech procedures concerning RPVI measures with other regulatory requirements that are based on the presumption of optimized materials. The Czech regulatory concept is being developed for existing NPPs that are not constructed and built fulfilling current state-of-the-art requirements. The basic requirement of state-of-the-art RPV integrity, the use of optimised steels (not radiation-sensitive) is not met for the Temelín RPVs.

During former Workshops with Czech Experts in the frame of the Trialogue and the project PN2 (High Energy Pipe Lines at the 28,8 m Level (AQG/WPNS country specific recommendation) [Item No.1]) information has been compiled concerning:

- Operational pressure-temperature limit curves (calculated in accordance with the Westinghouse methodology)
- First attempts of PTS analyses related to the DEGB (double-ended guillotine break) of the main steam line at the 28,8 m level
- Modifications of the WWER-1000 surveillance programmes related to the elimination of the evident deficiencies of these programmes
- Material properties of the RPV steel and weld material at NPP Temelín
- NDE (non-destructive evaluation) qualification approach

Specialists Workshop (PM3)

The preparatory activities for the workshop included the development of briefing material and the briefing for the Austrian delegation. The principles of PTS analysis requirements, surveillance programme implementation including known modifications compared to the former WWER-1000 surveillance programmes, the compliance and differences with the state-of-the-art practices as identified were described and commented with respect to their safety significance.

The Specialists' Workshop scheduled in the frame of the "Conclusions of the Melk Process and follow-up" for the first half of 2004 took place at SÚJB in Prague during May 24th/25th, 2004.

The Specialists' Workshop in Prague 2004 was concentrated on the performance of the PTS analyses and PTS related operational precautions. Other topics related to RPVI were not touched at the Workshop, such as, main coolant recirculation line penetrations, vessel head control rod penetrations, core instrumentation and other service penetrations, main flanges' tightness, and major environmental and other damage mechanisms contributing to the loss of integrity, like main coolant chemistry, hydrogen diffusion, corrosion, load cycling, severe accident behaviour, as well as integrity preservation and surveillance measures ascertaining LBB applicability and leakage detection instrumentation. During the Workshop the Czech side presented a set of 16 presentations, which are reflected in this report in the respective chapters.

In addition to the Workshop presentations Czech Experts delivered a second series of presentations on October 7, 2004 in Řež, in order to complement the information provided at the Workshop. The important clarifications provided there and the results of the discussions that followed the presentations have also contributed to this report.

Preparation of the Preliminary Monitoring Report (PMR) (PM4)

It was intended to publish a Preliminary Monitoring Report evaluating the Czech presentations during the Workshop in relation to the international practice and the Czech legal basis. The results of the bounding case calculations performed in support of the monitoring effort should have consolidated already the argumentation in the PMR [ANNEX D].

Given the tight schedule on the one hand and the need to consider additional information arising from the October Meeting and the Bilateral Meeting 2004 on the other hand, the Federal Ministry of Agriculture, Environment and Water Management finally decided to forgo the PMR.

Bilateral Meeting (PM5)

The 13th Bilateral Meeting under the Agreement between the Government of Austria and the Government of the Czech Republic on Issues of Common Interest in the Field of Nuclear Safety and Radiation Protection took place in Dolni Dunajovice, on 29-30 November 2004 On this occasion the preliminary results of the monitoring were presented to the Czech delegation and the replies were discussed.

An overview of the activities that would follow this Bilateral Meeting was given and further information on the issues associated with RPVI and PTS was envisaged to be treated in future Bilateral Meetings.

Final Monitoring Report (FMR) (PM6)

The evaluation of the additional information provided by the Czech Experts and of additional results of pilot studies conducted was incorporated into this Final Monitoring Report.

IV. Main findings

IV.1 Reactor pressure vessel integrity and PTS analyses

From monitoring Reactor Pressure Vessel Integrity issues of the Temelín NPPs as treated in the Czech Republic the Austrian Experts' Team has made the following findings:

A PTS analysis has to be performed according to Code regulations in any country before the start-up as part of the licensing to demonstrate the structural integrity of the RPV throughout the service life.

The NPP Temelín was started without performing a pre-service PTS analysis. The Regulatory Body accepted the operational limiting p-T curves (according to an analysis performed according the methodology of the Westinghouse concept) as preliminary demonstration of RPVI.

The Austrian Experts' Team did not consider the operational pressure-temperature limits (Westinghouse concept) as an appropriate substitute for a PTS analysis; furthermore the performed analysis was based on non-conservative assumptions[1]. It has to be recalled that – the Roadmap states with respect to the RPVI actions to be performed by the Czech side: "…a step by step approach with full respect of the IAEA Guidelines for the PTS analysis".

The Workshop presentation on first results of PTS analyses within the frame of the project PN2 (Conclusions of the Melk Process and Follow-up: Item No.1: High Energy Pipe-Lines at the 28,8 m Level) provided first information on the concept of PTSA being performed for NPP Temelín.

The state-of-the-art RPV integrity requirement to use optimised steels, that are not radiation embrittlement susceptible, is not met for the Temelín RPVs.

<u>Code regulations and state-of-the-art practice</u>

NPP Temelín construction was started with former Soviet support, according to the Soviet design and manufacturing regulations. Even during the late construction phase under former Czechoslovakian and later Czech Republic authorities the Russian Code regulations were the legal regulatory base.

[1] Please refer to the main report for details and the [ATPP 2001].

According to SÚJB (Workshop May 2004) the current Czech ruling on RPVI and PTSA is based on:

- Section IV if the Association of Mechanical Engineers of the Czech Republic Code (ASI Standard): Residual lifetime assessment of WWER nuclear power plants components and piping.
- The instructions and recommendations for lifetime assessment of WWER RPV and reactor internals during NPP operation [SUJB 1998]
- IAEA-Guidelines on PTSA for WWER nuclear power plants [IAEA 1997]
- The Czech Experts have been taking the lead in a EU research program called VERLIFE aiming at the development of the so-called VERLIFE methodology as non-mandatory guideline for the demonstration of WWER-RPVI. The VERLIFE methodology was approved by SÚJB in the beginning of May 2004. It should be noted however, that further development of the VERLIFE methodology is still ongoing. The overall concept of the VERLIFE methodology is in principle comparable to Western practices. Nevertheless, comprehensive documentation on the VERLIFE methodology is not available to the Austrian Experts.

The demonstration of reactor pressure vessel integrity can be performed in probabilistic or deterministic manner in principle. The various national requirements do not call for a particular way of demonstration, however many regulatory authorities have adopted the IAEA Guidelines as the basis.

In different National Codes there are requirements for safety factors to be used within the calculations, either within the fracture mechanical calculations (Russian Code) or with respect to the postulated crack sizes (German Code) or requirements related to specified conditions (French Code)[2].

The possibility to take credit of a possible WPS (warm pre-stress) effect reduces the inherent safety margin. This effect is respected as part of the new Russian Code; it was also applied in Germany. Although included in the ASME Code, it was never used in the U.S., and it is not included in the French Code. The IAEA Guidelines (1997) would allow for applying WPS[3].

Although the IAEA Guidelines on PTSA (1997) are part of the Czech licensing and are cited in the "Conclusions of the Melk Process and follow up" as the basis for the PTS analyses to be performed, the VERLIFE methodology has adopted no safety factors in the Stress Intensity Factors calculations (the IAEA Guidelines, however make use of safety factors in comparable cases). Furthermore, the WPS effect is assumed with 90% of the global maximum of the peak stress intensity factor (as compared to 80% of the peak level as defined in the IAEA Guidelines). This is a considerable reduction of conservatism in comparison with the recommendations of the IAEA Guidelines for PTSA.

ETE sponsored activities concerning PTS analyses

The PTSA methodology applied by NRI Řež for the NPP Temelín appears to be in accordance with the recommendations of the IAEA Guidelines and the international state-of-the-art using validated computer codes such as the RELAP5/mod.3.2 code for the general thermal-hydraulics, CATHARE and the engineering model based REMIX/NEWMIX codes for the mixing in the downcomer and the temperature fields at the RPV wall, and SYSTUS for the structural/ fracture mechanics analyses. For the calculation of the stress intensity factors both, the application of FEM computer codes or engineering knowledge based analytical approaches may be used. The new Russian code has included analytical approaches based on weight functions.

[2] Please refer to the main report for details on codes and their application.

[3] IAEA cautions that the application of WPS "should be carefully considered since it may not be fully applicable in the highly embrittled materials".

The application of the named codes was approved by SÚJB.

The Experts' Teams' evaluation of the Czech approach as described in the Workshop presentations takes into account also the results of the monitoring bounding case calculations. The resulting conclusions are summarised in V.1.

The demonstration of the RPV integrity is performed in terms of the safety margin between maximum allowable value of the materials critical brittle fracture temperature T_k^a and actual RPV material specific value T_k. The maximum allowable value of critical brittle fracture temperature is derived from the fracture mechanics calculations of the load paths for each postulated defect in combination with every selected accident transient.

The applicability of the WPS effect is still controversial in the international community due to theoretical and experimental uncertainties. The actual RPV material state is described by the fracture toughness curve where T_k is determined by the formula from the Russian Code using the specified embrittlement coefficient. The experimental data from the surveillance programme are supposed to confirm the conservatism of the embrittlement coefficient.

Even the actual situation with transients used for PTS analyses, which do not include all worst-case conditions and neglecting the safety factors the Czech analyses results for the maximum allowable critical temperature of brittleness T_k^a are extremely close to the EOL values of the T_k for WWER-1000 materials. These reactors experience extreme PTS conditions since they can compensate a Double-End Guillotine Break of the main recirculation line with their increased emergency cooling systems injection capabilities of cold water into the downcomer during emergency operation. This emergency cooling has three consequences important for PTS:

1. Reactor pressure vessel wall inner surface layers are cooled down very rapidly to the temperature range 80 ÷ 20 [°C].
2. The high capability of the emergency cooling systems causes cold plumes in the downcomer and steep temperature gradients over the pressure vessel wall thickness.
3. For small and intermediate breaks the fast compensation of the coolant loss by the high flow rate from the emergency cooling systems causes an early and rapid re-pressurisation of the primary circuit, which adds to the thermal shock load.

In case internationally recommended safety factors would be considered in the Stress Intensity Factors (SIF) calculations for Temelín, the critical embrittlement conditions would occur significantly before the projected End of Life (EOL) of the Temelín NPP. These results confirm the unfavourable situation of WWER-1000 RPV also revealed by PTSA results. Since the most critical transients have not yet been analysed, the situation might be even worse.

In general, the omission of safety factors in RPV design becoming accepted practice would result in a significant reduction of safety margins.

IV.2 Temelín activities concerning material embrittlement monitoring (surveillance programme) and material properties

Surveillance programme

In principle, the material degradation due to neutron irradiation can be predicted based on the knowledge of tests on RPV steel specimens performed over the years during development of the specific type of reactors. The results of this broad experimental background have been used for the definition of material requirements in the National Codes. Besides this in practically all countries the material degradation (embrittlement) of the plant specific RPV materials (except for the very first NPPs) is monitored throughout the operational lifetime by executing the so-called surveillance programmes.

Surveillance programmes require representative samples of the vessel material (representative samples are made using oversize cuts of the ring base material and special welds, manufactured using identical base material and weld electrodes, and identical manufacturing conditions as for the RPV). The irradiation capsules for the surveillance samples have to be located in the RPV so that the neutron flux at the sample location is higher than at the vessel wall in the belt region in order to reach an accelerated irradiation that allows prognostic information on the embrittlement behaviour. The so-called "lead factor" is the ratio of the flux at the surveillance sample position relative to the flux the vessel wall is exposed to in the region of the active core.

This procedure provides an accelerated irradiation condition for each specimen set that allows predictive data on the embrittlement with a need to restrict the lead factor to about 2 to avoid distortion due to high dose rate. The capsules are withdrawn after regular time periods for destructive testing of the material specimens.

The original surveillance programmes for WWER-1000 NPP had severe deficiencies and this fact has been confirmed by two TACIS projects:

"In the framework of these projects, the validity and representatively of WWER-1000 surveillance data and other experimental results have been done. But due to the low fluence value and insufficient number of surveillance specimens the accuracy of radiation embrittlement assessment of RPVs was not high. It was also confirmed that the specimen temperature was possibly higher than the vessel wall temperature. In this case the surveillance results for vessel embrittlement assessment may give non-conservative forecast." [KRYUKOV 2000]

The modification of the surveillance program for WWER-1000 implemented at NPP Temelín has eliminated the obvious deficiencies of the original WWER-1000 surveillance programmes, with respect to irradiation temperature, neutron flux and fluence at the sample location. The embrittlement information of the Temelín irradiated samples will therefore provide the first reliable data on WWER RPV material embrittlement.

Temelín RPV material properties

The Czech determination of the critical temperature of brittleness T_k is defined and performed in according to Russian Code regulations, which are quite close to the Western practice. The shift of this temperature caused by neutron embrittlement is also performed and defined according to Russian Code regulations.

According to the information available to the Austrian specialists the (unirradiated materials') initial critical temperature of brittleness T_{k0} is highest for weld no.4 in both units. Although the neutron flux in weld no.4 is about 80% of the neutron flux at weld no.3, the weld material at the location of the weld no.4 has to be considered leading with respect to neutron embrittlement.

The predictive values for neutron embrittlement in the Russian Code are based on experimental test results on the WWER-1000 steel irradiated 15Ch2MNFAA 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 Řež using original RPV materials of the NPP Temelín unit-1 have also been realized with high lead factors (about 160 or higher).

This means that the possible dose rate effect (higher embrittlement at lower neutron flux compared to high neutron flux for identical fluence) could have affected the results[4].

[4] A question of the Austrian Experts during the Workshop 2004 on the possible influence of the dose rate effect was answered in the sense that this effect cannot be excluded and will be studied in future research programmes.

The first capsule with irradiated ETE-samples has been withdrawn during May 2004; the evaluated data will be available one year thereafter. Until that time the PTS analysis performed is using the potentially non-conservative predictive values of the Russian Code. The VERLIFE methodology used by the operator does not foresee the use of any safety margin covering these uncertainties. The Russian Code and the IAEA Guidelines require a safety margin of ΔT =10 [K] with respect to the uncertainties of critical temperature of brittleness. The U.S. regulations require the use of a safety margin to cover the uncertainties of the experimental method for the determination of the initial RT_{NDT}[5] and the uncertainties of the determination of ΔT_{RTNDT} (15,5 [K] for welds and 9,5 [K]). However, other National Codes do not provide rules for the use of safety margins to consider the uncertainties.

Despite the fact that the surveillance data will be as reliable as required, it has to be stated, that each removal of capsules will contribute with only one single value to the embrittlement versus fluence (operation time) behaviour representation. Evidence is there, that this surveillance programme during service life cannot establish the statistical basis required for reliable prediction of the embrittlement behaviour. Since many publications indicate that the values of embrittlement for WWER-material specifications as defined in the Russian Code might be non-conservative, these uncertainties should be considered in the discussion of the safety margins against brittle fracture of the RPV material in case of PTS events.

Fracture toughness curve for PTSA

The assessment of the structural integrity or the residual lifetime of an RPV by a PTS analysis includes the comparison of the calculated load path in case of a PTS transient and the actual material state of the RPV steel which degrades mainly due to neutron embrittlement. This material state is described by the fracture toughness as a function of temperature $K_{Ic}(T-T_k)$.

The formula defined within the VERLIFE methodology can be considered to be the most conservative when compared to the Russian Code and the ASME Code[6] (which is identical to the French and the German Code). It should be noted that the formula is equivalent to the fracture toughness curve recommended in the IAEA Guidelines.

However, the static fracture toughness data shown during the Specialists' Workshop in Prague 2004 did not demonstrate that the used fracture toughness curve could be considered to be conservative for irradiated WWER-440 materials. There is also no evidence that the static fracture toughness data from WWER-1000 materials will be described conservatively by the used fracture toughness curve – sort of a master-curve.

IV.3 NDT (Non-Destructive Testing) programme

Nuclear reactor pressure vessels are submitted at regular intervals to in-service inspections (ISI) in order to detect and monitor flaws. Since detection methods are improving it is possible to detect fabrication flaws only during in-service inspection – it is also possible that flaws grow during service reaching a detectable size.

NDT programs have to be qualified using specific test samples that are representative for the components to be inspected.

The qualification procedure at the test sample KB 190 for the RPV-wall inspection at Temelín NPP has obviously only been finalized very recently. This indicates that qualified inspection results available up to now are based only on a limited number of wall inspections that conform to the qualified and accepted procedure.

[5] Reference temperature for the ductile brittle transition temperature in the Western terminology is comparable to the critical temperature of brittleness in the Russian Code.

[6] Below about 70 [MPa$\sqrt{m}$] the ASME curve is slightly more conservative.

It is not clear whether a complete zero-NDT map exists. The comparability of any available inspection results with the qualified methods has not been demonstrated yet.

This issue of NDT will be treated in detail in project PN10: Integrity of Primary Loop Components – Non Destructive Testing (NDT) [Item No. 4][7].

IV.4 Core design – fluence management

Fluence estimates calculated at the RPV wall are very sensitive to the calculation procedures. Because of high neutron fluence attenuation between the core and the RPV wall in the core region the calculated RPV fluence is also strongly sensitive to the physical model of the core and RPV internals as well as to the mathematical model for the neutron transport calculations. The accurate determination of the RPV fluence is difficult and comparisons of measured and calculated data show a varying degree of agreement for different WWER designs and different core loading schemes.

In ETE Westinghouse implemented a new core concept replacing the original concept of the Russian designer. It is not known to what extent this concept has been validated. Since it is a sort of prototype assembly – until the construction of ETE there has been no essential core modification with respect to the original Russian design – one should assume that there was an extensive validation process.

IV.5 EOPs

During the Specialists' Workshop in Prague, a presentation was provided on how the Temelín Emergency Operating Procedures (EOPs) address PTS conditions. The state-of-the-art in procedural aspects of pressurized thermal shock (PTS) is to have symptom-based Emergency Operating Procedures (EOPs) in place to identify and manage potential PTS conditions and bring the plant to safe shutdown without reactor coolant system pressure boundary failure, with adequate core cooling at the same time. Should conditions occur nonetheless which give rise to core damage, Severe Accident Management Guidelines (SAMGs) are required to be available to limit core damage and mitigate the consequences of such a core damage. Symptom-based EOPs, the result of a joint CEZ-Westinghouse project, were implemented at Temelín in 1998. Similarly, SAMGs are in the stage of implementation to be completed by the end of 2004. ČEZ has adopted a standard and well-recognized procedural approach to managing PTS events in implementing Westinghouse EOPs and SAMGs.

This issue has been treated in detail within the project PN7 Severe Accidents Related Issues – [Item No. 7b].

IV.6 Training programmes – QA programmes

Implementation of an extended program like RPVI assurance, including the VERLIFE concept, involves many disciplines and TSOs. Therefore extended collateral programs have to be set up for training and quality assurance before and during the time period the program is enacted – in this case the operational life of the plant.

The information about the personnel training program related to PTS events provides for a satisfactory picture. The activities correspond with comparable European situations.

The information concerning the QA program was about some general information concerning QA procedures and acceptance criteria imposed by SÚJB for computational analyses applied at NRI.

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[7] The Items are related to ANNEX I of the "Conclusions of the Melk Process and Follow-up".

IV.7 SÚJB position

Expectations about the involvement of SÚJB were largely clarified during the Workshop. The substance of the decisions bases for e.g. adopting the VERLIFE methodology has not been discussed. The schedule for implementation of PTS related RPVI measures was also not discussed.

At the Specialists` Workshop all the topics of interest were addressed in a general manner and specific information was obtained regarding certain questions. The Experts' Team received insight into the essential topic of external support and independence, which is also addressed in the NRA fundamentals.

The IAEA IRRT Mission to the Czech Republic [IRRT 2000, paragraph 1.7.1] recommended, inter alii, to address external and independent expertise. It stated that SÚJB's personnel capacity and possibilities should be increased by all means appropriate.

Generally and specifically this should be the case with respect to the RPVI and PTS issues at the Temelín NPP in a sufficient and efficient manner.

V. Conclusions

The demonstration of RPVI (reactor pressure vessel integrity) throughout service life is performed by the Czech Experts, for Temelín NPP, using the VERLIFE methodology. From a comparison to the Russian Code and the IAEA Guidelines, the Austrian Experts' Team has found that the VERLIFE methodology, as applied to the Temelín RPVs, makes use of reduced safety margins (i.e. reduction of the postulated crack size, elimination/reduction of safety factors, non-conservative assumptions for the fracture mechanics analyses). In combination with other uncertainties concerning material/embrittlement properties and apparent reductions of conservatisms in several respects, the Austrian Experts' Team considers the resulting global safety margin for the Temelín RPVs as not being sufficient.

The complete VERLIFE methodology requirements and their application to the Temelín NPP have not been available to the Austrian Expert's Team. For the applied VERLIFE methodology the Austrian Experts' Team had to rely essentially on the information provided during the Specialists´ Workshops.

The Austrian Experts' Team also found that the Czech approach – as presented – for PTS analyses is in accordance with the state of science and technology, with respect to the concept, the methodology and the applied computer codes. The most severe transients analysed are well comparable to those regarded as representative for WWER-1000 installations according to current knowledge. All accident groups important in a PTS analysis were considered.

However, a number of issues remain to be clarified:

- The basis for the analyses appears to be insufficient: Although all accident groups important in a PTSA were analysed, in some cases the time frame of the simulation might not have caught critical loads to the reactor pressure vessel, since simulation results were available only for the phase ending just before repressurization would take place. Within a number of accident groups, the transients analysed in some cases cannot be considered as the most critical ones. For some transients it is necessary that emergency operating procedures be performed within a narrow time window to avoid brittle failure of the RPV.

- There are apparent reductions of conservatisms. Some VERLIFE criteria are weaker than those required by the IAEA Guidelines. Applying the values concerning postulated crack sizes, safety factors, WPS (warm prestressing effect criterion) as required by the IAEA Guidelines would not result in the demonstration of RPVI requirements' fulfilment throughout lifetime.
- Uncertainties – procedural as well as intrinsic – identified regarding the PTS assessment for Temelín NPP concern, for example: TH transient models, mixing behaviour models, embrittlement behaviour of the RPV materials as well as initial materials' brittleness properties, fluence determination and the introduction of measures for fluence minimization, and areas of in-service-inspection (ISI), where qualification has not yet been achieved. These are further critical points remaining for clarification.
- Conservatism is further reduced by including the intact cladding zone as structural reinforcement into the Finite Element model, including non-conservative assumptions for fracture mechanics analyses at the cladding/ferritic steel interface (as confirmed by a pilot study of the Austrian Experts' Team). Accordingly, not all types of underclad cracks have been evaluated.

Regarding the surveillance program, which is monitoring embrittlement progress, in particular the location of the samples, it has to be pointed out, however, that it represents a considerable improvement compared to other WWER-1000s of the same vintage.

Consequently, the future exchange of information on RPVI and PTS should above all cover the following issues:

- Regarding PTS analyses, the consequences of additional critical conditions, and of an extended time frame for some of the sequences calculated, are of interest, as well as the consideration of all crack sizes and crack positions of relevance in fracture mechanics (including stability considerations).
- The progression of embrittlement and the remedies taken should be further observed. This includes surveillance results for both units of the Temelín NPP, in particular the results of samples with higher initial critical temperatures brittleness, irradiated in unit 2.
- The comparison of materials' characteristics determined within the qualification tests, the extended acceptance tests and the lifetime evaluation programme with the surveillance programme data is of interest, in order to evaluate the scatter of materials' properties.
- Embrittlement mitigations measures, in particular core configuration, refuelling pattern and enrichment changes, are of interest.

In the course of further information exchange, the issues listed here could be combined with the issues remaining for information exchange under Item 4 (Non Destructive Testing) of ANNEX I of the "Brussels Agreement", regarding the reliable detection of all PTS relevant defects.

Detailed conclusions

Benchmarking the presentations at the Specialists' Workshop against internationally accepted guidance, recommendations and ruling has led the Austrian Experts' Team to the following observations: Many of these observations are also based on generic calculations and investigations that were conducted while preparing the workshop.

- The Austrian Experts appreciate that the Czech side is no more considering the operational pressure-temperature limit curves as appropriate demonstration of avoidance of unacceptable PTS sequences.
- The RPVI concept, as it pertains to the PTS analysis approach, appears to follow the state-of-the-art practice and the IAEA Guidelines with respect to analytical methodology. The IAEA Guidelines safety precautions were significantly reduced the way they are interpreted in the new VERLIFE methodology.

- The presented Czech approach for PTS analyses (part of the VERLIFE) with respect to the concept, the methodology and the applied computer codes are considered to be in accordance with state-of-the-art procedures.
- Evidently all thermal hydraulic calculations work has been performed with state-of-technology computer codes, which were validated for WWER-1000 use. Once completed the RPVI/PTS related TH-analyses can be considered comprehensive. TH-analyses should provide a sound basis for the selection of candidate transients for the mixing and heat transfer calculations to be conducted subsequently. The use of assumptions, which are not conservative for the specific scope and represent therefore an impact on safety, should be reconsidered.
- The most severe transients are by all means comparable to those considered representative for WWER-1000 installations according to current knowledge. In some instances the time frame observed in the simulation might not have caught the essence of the loading to the RPV, since re-pressurization during the up-following accident-sequence might just not have taken place that early (i.e. before ceasing the simulation).

With regard to "Mixing Calculations and Heat Transfer" issues:

- The mixing calculations for the accident transients within the PTS analyses performed appear to be in accordance with the state-of-the-art in international practice and comparable to calculations for other WWER-1000 reactors.

With regard to FEM calculations and Fracture Mechanics evaluation:

- The applied computer codes for the FEM simulation and the consideration of elastic-plastic material behaviour is considered to be in accordance with the actual state-of-the-art. The PTS assessment can be considered a consolidated approach, up to now unprecedented for WWER-1000 reactors.
- The IAEA Guidelines allow the use of postulated crack depths shallower than the normally required ¼ of wall thickness (which is for the WWER-1000 about 50 [mm]) for the case of the NDT-Program enabling the safe detection of the respective small defect sizes. For this case the IAEA Guidelines require the mandatory use of safety factors: Safety factor 2 for the crack depth or safety factor $\sqrt{2}$ for the stress and $\Delta T = 10$ [K] for the embrittlement induced shift of the critical brittle fracture temperature. In accordance with VERLIFE [PISTORA 2004a] the Czech Experts postulate a crack depth of 20 [mm] only ($^{1}/_{10}$ wall thickness, which is significantly smaller than ¼ wall thickness) but do not apply any of the safety factors. (e.g. as required according to the IAEA Guidelines).
- The Czech approach is also deviating from the IAEA Guidelines [IAEA 1997] with respect to the missing variations of the crack size and crack geometry. The following investigations have not been presented:
  - The analyses for very shallow cracks (a < 6 [mm]) and
  - Large cracks (a = 20 [mm] up to ¼ of the wall thickness) and
  - The variation of the aspect ratio to a:c = 1:10.
- The approach taken for integrating the cladding zone into the FE modelling introduces furthermore a reduction of conservatism, not only when excluding elliptical under-clad cracks, but also because assuming a Stress Intensity Factor (SIF) levelling out to SIF=0 exactly at the cladding/base-material interface does not correspond to reality. This has been reconfirmed by pilot case simulations conducted during the monitoring process.
- The FEM model represents one half of the reactor pressure vessel. This procedure does not include the stresses from the superposition of the cold plumes, the strain induced distortion of the cylinder and the interaction with the RPV bottom and the RPV head (deformation hindering). It should be noted, that this approach is in accordance with the international practice. The simulation using a mesh covering the complete RPV would represent an outstanding effort.

With respect to the PTSA:

- All accident groups important to be treated in a PTSA were analysed. For WWER-1000 reactors this is the first PTSA with a completeness not achieved up to now.
- The PTS loads for WWER-1000 are extremely high. For a postulated crack depth of only 20 [mm] the resulting T_k^a values are below 70 [°C] in four cases and 3 accident groups, no comparable behaviour is found with any other reactor types, e.g. WWER-440. This is a consequence of the very effective emergency coolant injection systems that are able to compensate large breaks up to ND 850 but induce at the same time a severe thermal shock load at the RPV wall.
- The lowest T_k^a values are found for small to intermediate break sizes, where in addition to the thermal shock load a full or partial re-pressurisation of the primary coolant circuit might occur .
- The operator must perform the appropriate emergency operation procedures (EOPs) at the correct moment in order to cope with several accident transients (PSV41) and at the same time avoid brittle failure of the RPV. However, it is not international practice to require “guaranteed” operational procedures of the personnel; therefore this must be considered a considerable reduction of conservatism in the handling of emergencies.
- Some accidents (PSV43) have not been calculated until to the point of applicability of the 90% WPS-criterion.
- In some cases the definition of the accident transients cannot be considered the most critical one: In the accident group PRZ SV the total loss of off-site power has not been included, although this is required by the IAEA Guidelines. Including total loss of off-site power would induce a re-pressurisation in the primary circuit following the re-closure of the pressuriser safety valve.

With respect to the safety factors required by the IAEA Guidelines it has to be stated:

- The VERLIFE methodology as applied by the Czech Experts for the Temelín NPP uses only postulated crack depths of 20 [mm] ($^1/_{10}$ wall thickness, which is significantly smaller than ¼ wall thickness) and no safety factors, which is not in line with the pertinent IAEA Guidelines.
- The VERLIFE methodology as applied by the Czech Experts for NPP Temelín is applying the 90% WPS criterion although the IAEA Guidelines recommend the 80% level, if applied at all. This modification significantly reduces further conservatism, which violates the need to compensate for uncertainties in embrittlement prediction for radiation-sensitive RPV steels.
- Even though applicability of the WPS effect is still judged controversial in the international community due to theoretical and experimental uncertainties[8] it is applied for the Temelín RPV integrity.
- The consequent application of the IAEA guidelines would lead to a different assessment result than advocated by the operator of Temelín, e.g. in cases where the 80% WPS-criterion, together with the safety factor $\sqrt{2}$ and the required safety factor ΔT = 10 [K] should be applied.

With respect to the surveillance program in the NPP Temelín and ETE RPV material embrittlement:

- The use of optimised steels – not radiation embrittlement susceptible/sensitive – one basic element of state-of-the-art RPV integrity, is not met for the barrier – the Temelín RPVs.

[8] This is the case especially with respect to the real situation in the component and the temperature/pressure history during a realistic PTS event.

- The modified surveillance program in the NPP Temelín allows the determination of reliable embrittlement data with respect to irradiation temperature and neutron flux/fluence at the samples irradiation location.
- The modified surveillance program causes inaccessibility of RPV wall in the container area and therefore for NDT in regions close to weldment 4, the active core and core zone.
- The evaluation of published surveillance results from WWER-1000 materials taking into account the estimated irradiation temperatures does result in considerable uncertainties about the neutron embrittlement of WWER-1000 steel. It is therefore obvious that the specification in the Russian Code (A_F = 20 for welds, 23 for base material) cannot be considered conservative.
- Although the first reliable results ever regarding a WWER-1000 will be available from the Temelín surveillance program, uncertainties about the WWER-1000 RPV steel embrittlement persist: the RPV specific surveillance program cannot provide a reliable statistics background for the prediction of the material degradation, since every set of samples withdrawn and evaluated provides for only one single data point to be added to the irradiation embrittlement versus time correlation.
- The embrittlement coefficients determined so far for Temelín specific materials are based on irradiation in test reactors with high lead factors. The existing dose rate effect might have adversely affected the embrittlement and the coefficients determined, i.e. the embrittlement might in reality be more advanced than the measured values indicate.
- The material properties data in the passports indicate that the initial critical temperature of brittleness T_{k0} can vary by tens of degrees from one weld metal charge to another. It has not been possible to check whether the temperature margin δT_M (10 [K] for the base material and 16 [K] for the weld metals) as defined within the VERLIFE methodology, in order to cover the scatter of the mechanical property values, have been taken into account for T_{k0} assessment.
- This fact and the uncertainties of the specified embrittlement coefficients need to be taken into consideration by using the safety factor ΔT as required by the IAEA Guidelines [IAEA 1997].
- Weld no. 4 in ETE-1 was welded with two different electrode heat charges (Sv12Ch2N2MAA, heat number 17084 and 170007) for both heat numbers surveillance samples were fabricated; the surveillance program of ETE-1 is performed using the samples welded with the same electrode heat than weld no. 3 (T_{k0} = -50 [°C]). The other weld metal with T_{k0} = –30 [°C] will be irradiated within the surveillance program of ETE-2. In view of the Austrian specialists this is a shortcoming because the results on irradiation embrittlement for the weld material with the highest T_{k0} of ETE-1 will not be available without significant delay.
- The fracture toughness curve formula used in the VERLIFE methodology can be considered conservative as compared with fracture toughness curves of other National Codes.

With respect to the NDT/ISI program performed in NPP Temelín:

- The ISI using ultrasonic NDT methods for the RPV cylindrical wall has successfully been qualified. The methods can as such be regarded to basically enable detection of all kinds of crack-like defects, which are of special concern for the PTS events and their analyses, e.g. cracks close to the claddings' interface to the base material layer with an $^a/_c$ aspect-ratio of e.g. 0,3 and with different depth, depending on the PTSA defects as postulated. A semi-elliptical crack seems to be the most critical for NDT, which starts at the cladding interface and extends 8 [mm] deep into the ferritic wall. Although qualification using the RPV wall test block demonstrated the basic potential of the applied UT methods to allow detection of those defects, some problems are not yet finally solved.

- The test block does not contain the cladding condition at the welds and on its vicinity, where one has to take into account a considerably higher noise level and therefore a higher false call rate, as mentioned in the qualification report. This requires special countermeasures, e.g. additional Eddy Current Testing (ECT) in areas with an elevated number of UT indications. This is particularly needed, because the VERLIFE concept requires an intact cladding, especially at locations of near cladding cracks in the ferritic wall. The remaining ligament between the crack tip and the wet inner surface can be proven with properly qualified ECT methods only. The safety evaluation regarding the absence of flaws important to PTS has not been finalized up to now, since neither the qualification of, nor the inspection using the ECT method as required has been carried out yet.
- Two more ISI areas bearing specific PTS concerns are the inner corner of the inlet nozzles and the welds connecting the primary loop to the RPV. For both areas qualification exercises have been announced, but have not been finished yet and presented. Of special interest are the PTS relevant crack sizes within the nozzle corner and the connecting weld, in order to judge the difficulties the NDT techniques will have to guarantee sufficient detectability and a reasonable false call rate.
- In view of the not yet finished remaining NDT activities, but needed to prove the absence of all kind of PTS relevant cracks, one must conclude, that the NDT inspections carried out until today cover only in part all the ISIs required. According to the information given at the PN9 Workshop, completion of the ISI concerning the PTS analysis is in preparation, with several qualification activities ongoing, but will certainly not be reached before the foreseeable next RPV ISI.

With regard to Core Design and Radiation Embrittlement Mitigation:

- The OUT-IN strategy is a well-known early means of embrittlement mitigation; the ETE specific information contained in the presentation did not give a clue to the question, whether introduction is made for irradiation embrittlement mitigation, or just as a side effect of power output optimization. The PTS relevant effects of the RPV fluence reduction management can be derived from the fluence distribution only. Nevertheless, information presented was limited to power distribution sketches.
- The statement during the Specialists' Workshop, that operation will take place well below fluence calculation input, does not per se endorse that embrittlement is managed properly. The RPV fluence reduction management policy is one element to be enacted along with plant operation.
- The Westinghouse core design used in a WWER-1000 reactor is the first of its kind to be validated. Apparently, the core design has not yet been modified aiming to a fluence minimization at the reactor pressure vessel wall in order to reduce the neutron embrittlement of the steel. This improvement is envisaged be implemented at one of the upcoming refuelling outages of the core. Up to now the intended changes have not been presented.

With regard to EOPs and SAMGs transition:

- Extensive feedback from plant analyses was used to more appropriately adapt the EOPs outline and elements to an up-to-date emergency management tool. It can be understood from the overview presentation, that the concept is suitable for proper adaptation. This work is evidently a successfully ongoing process.
- The EOPs as well as the SAMGs and associated measures are well in line with the state of science and technology requirements, given the equipment to be used be qualified or been qualified for the intended use in the respective operational regime.

Conclusions concerning the issue of quality assurance and training:

- Due to the unavailability of detailed information it is not possible to judge the efficiency of quality assurance programmes related to RPVI activities at NPP Temelín. In any case, together with the evaluation of quality assurance the improvements achieved for QA are appreciated.
- Verification and consolidation of a sound understanding of the actual RPV and plant systems situation requires procedures and management structures to be set up. This management should be set up for a process that is supposed to last for the entire plant life. The related prerequisites have been set-up in adequate proportions.
- The training and implementation activities are comprehensive and compare well with activities in other NPPs in Europe. In some instances thoroughness was most probably given precedence before timeliness when implementing EOPs training opportunities.

Conclusions concerning the SÚJB position:

- The SÚJB position on the “PTS requirements” implementation versus the licensee is an indication of their observing position in assuring the RPVI and PTS precautions fulfilment.
- In line with the IAEA IRRT Mission recommendation the Experts’ Team considers that it is a valid aim to enhance SÚJB´s “strength”. Its personnel capacity and possibilities ought to be increased by all means appropriate and necessary also in the RPVI and PTS context.

VI. PTS – items of further interest

The team of Experts recommends pursuing topics of high priority in the framework of the pertinent Bilateral Agreement between the Federal Republic of Austria and the Czech Republic. This concerns the implementation and results from the RPVI Program, VERLIFE and the related PTSA. In addition, since the ongoing RPVI/PTS information exchange process is supposed to be continued for the entire plant life, it is recommended to follow plant operation by continuous exchange of information.

Since the present RPVI work did not explicitly take into consideration cold over-pressurisation and outage-issues, no comments will be found here on these topics.

These items recommended are as follows:

- The consideration of additional critical conditions, such as total loss of off-site power,
- The time frame of sequences calculated – some transients’ simulations have not been conducted up to a time-span sufficiently long, that any over-pressurisation during the left out accident transient could have been captured – and
- The consideration of fracture mechanics regarding all the crack sizes and crack positions of relevance, and stability considerations (smaller cracks might grow and become instable during the up following transient sequences).
- The embrittlement progression as well as the remedies taken and the actual RPVI verification and consequences.
- The comparison of the materials characteristics determined within the qualification tests, the extended acceptance tests and the lifetime evaluation programme cited during the Workshop [BRUMOVSKY 2004a] with the surveillance programme data in order to evaluate the scatter of materials characteristics.
- The information on the results of the surveillance programme for both units. Special emphasis should be dedicated to the surveillance results of the weld no.4 samples (including the heat affected zone). The first results of the surveillance capsule removed in May 2004 will be available in 2005.

- The information on the results of the surveillance samples irradiated in unit 2 (esp. specimens of weld no.4/unit-1 and weld no.4/unit2, including HAZ) should be included in the future information exchange with special emphasis during the next years. At the same time it would be desirable to obtain information whether specimen of weld number 2 are included in the PTS considerations.
- Continuous information on the experimental assessment evaluation of the neutron embrittlement of ETE materials, using surveillance specimens, in order to confirm the application of temperature margins as defined in the VERLIFE methodology (upper boundary of the radiation induced T_k shifts to be used in the RPV lifetime evaluation).
- The Temelín RPV embrittlement mitigation is of utmost importance for RPVI; therefore fuel-reload as well as reload-pattern changes are envisaged after one of the next campaigns. The information provided up to now is coarse; it stipulates further interest.

Future information exchange should also include:

- Main coolant recirculation line penetrations,
- Vessel head control rod penetrations,
- Core instrumentation and other service penetrations,
- Main flanges' tightness, and
- Major environmental and other damage mechanisms contributing to the loss of integrity, like main coolant chemistry, hydrogen diffusion, corrosion, load cycling, severe accident behaviour, as well as integrity preservation and surveillance measures ascertaining LBB applicability and leakage detection instrumentation,
- The damage progression as well as the remedies taken and the actual RPVI verification and consequences,

since the Workshop did not cover those RPVI relevant issues.

Concluding statement

The Czech Experts make use of the VERLIFE methodology for demonstrating RPVI (reactor pressure vessel integrity) throughout service life of the Temelín RPVs. Compared to the Russian Code and the IAEA Guidelines the VERLIFE methodology has reduced the safety margins, adopted via inherent methodologies like the reduction of the postulated crack size, reduction of safety factors, the non-conservative fracture mechanics assumptions etc.

In combination with other uncertainties, such as modelling of TH transients, mixing behaviour modelling assumptions, material and embrittlement properties, fluence determination, NDE reliability, etc., the resulting global safety margin cannot be considered sufficient. Therefore the Austrian Experts' Team recommends continuing to follow up on those items, relevant for the completion of the VERLIFE methodology:

- Additional PTS analyses and their upgrading
- Surveillance specimen evaluation (of both units)
- Integrity verification dedicated NDE program
- Progress in embrittlement mitigation

The update of the Temelín RPVI demonstration specification based on the VERLIFE methodology would also be of high priority.

ZUSAMMENFASSUNG

I. Grundlage und der Hintergrund für das Projekt

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

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

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

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

Das Österreichische Bundesministerium für Land- und Forstwirtschaft, Umwelt und Wasserwirtschaft beauftragte das Umweltbundesamt mit der Gesamtkoordination der Umsetzung der „Roadmap“. Jeder Punkt in der „Roadmap“ entspricht einem spezifischen technischen Projekt.

Das hier behandelte „Roadmap“ Projekt betrifft den Informationsaustausch zu:

Punkt Nr. 3 – Reaktordruckbehälterintegrität und Schockbelastung unter Temperatur und Druck (das ExpertInnen-Treffen war für die zweite Hälfte 2004 vorgesehen): *„Dieses thematische Treffen wird den Status der PTS – Analyse (Pressurized Thermal Shock) behandeln.“*

Inhalt des „Roadmap“-Prozesses in diesem Projekt entsprechend der Feststellung im ANHANG I des „Brüsseler Abkommens“:

„Die Integrität des Reaktordruckbehälters unter Schockbelastung durch Temperatur und Druck ist mit einer hinreichenden Sicherheitstoleranz gegen Sprödbruch während der gesamten Lebensdauer des KKW aufrechtzuerhalten.“

Zusätzlich beinhaltet ANHANG I Aussagen zu dem „gegenwärtigen Stand und spezifischen geplanten Aktionen“:

„Das KKW Temelín wird unter Beachtung der entsprechenden betrieblichen Druck-Temperatur (p-T) Grenzkurven, die mit Hilfe der Westinghouse-Methodik berechnet wurden, genehmigt und betrieben. Diese Berechnungen werden durch eine Reihe zusätzlicher PTS-Analysen für beide Blöcke erweitert, wobei ein Schritt-für-Schritt-Verfahren unter vollständiger Beachtung der IAEA Guidelines für die PTS-Analyse angewendet wird. Die PTS-Analyse wird entsprechend einem genehmigten Projektplan für diesen Punkt abgearbeitet.“

Im Namen der Österreichischen Regierung hat das Umweltbundesamt ein internationales ExpertInnen-Team mit dem technischen Support zur Überwachung der Implementierung der RPVI-PTS-Thematik auf technischer Ebene beauftragt, wie im ANHANG I der „Schlussfolgerungen des Melker Prozesses und des Follow-up“ aufgezeigt wird. Dieses spezielle technische Projekt wird als PN9-Projekt bezeichnet, welches insgesamt sieben vorgegebene „Projektmeilensteine“ (PM) umfasst.

Ein ExpertInnen-Treffen (Specialists' Workshop) zum Roadmap-Punkt 3 Reaktordruckbehälterintegrität und Sprödbruchsicherheit unter Thermoschock fand in Prag, am 24. und 25. Mai 2004 statt, entsprechend Artikel 7 (4) des Bilateralen Abkommens über den Informationsaustausch zur Nuklearen Sicherheit. Ergänzt wurden die ExpertInnen Treffen-Ausführungen später, am 7. Oktober 2004 durch Vorträge in Řež, die zusätzliche Detailinformationen und Antworten zu einzelnen Fragen lieferten. Die beiden ExpertInnen-Treffen waren Schlüsselelemente in dem Monitoring-Prozess. Die Auswertung der während der ExpertInnen-Workshops erhaltenen Informationen durch die ExpertInnen und das Team spielt die tragende Rolle für die Ausarbeitung des „Final Monitoring Report“ durch das österreichische ExpertInnen-Team.

Die technische Unterstützung bei der Überprüfung der Umsetzung der „Schlussfolgerungen aus dem Melk-Prozeß und Follow-up“ war in technologischer Hinsicht für die Reaktordruckbehälterintegrität und Sprödbruchsicherheit unter Thermoschock auf die Auswertung der Frage konzentriert, wie die methodische Umsetzung durch die tschechische Seite (Betreiber und Aufsichtsbehörde) hinsichtlich der tatsächlichen Durchführung erfolgt. Speziell war vorgesehen, sich auf die Untersuchung der Umsetzung des Überwachungsprogramms und die Durchführung einer umfassenden PTS-Analyse zu konzentrieren, einerseits auf der Grundlage der in der EU weitgehend üblichen Praxis, und andrerseits bezogen auch auf neuere Entwicklungen der Erkenntnisse zu WWER-Anlagen, sowohl in technischer Hinsicht, wie auch in Hinsicht auf die Genehmigung.

II. Zielrichtung und Aufgabe des Projektes PN9

Für das Kernkraftwerk Temelín, ursprünglich Sowjetischer Bauart, später mit westlichen Sicherheitskonzepten und westlicher Technik aufgerüstet, wurden erst spät in der Errichtungsphase Sprödbruchsicherheitsanalysen (PTSA) unter Thermoschockbedingungen (PTS) und RDB-Integrität behandelt. Die Russische und Westliche Normen verlangen jedenfalls eine PTSA vor Inbetriebnahme. Während der ExpertInnen-Treffens im Rahmen des Melker Abkommens stellte sich heraus, dass der Prozess der PTS-Vorbeugung in Temelín erst spät erfolgte und unvollständig war. Die Informationslage über Details des Konzeptes am KKW war zudem nicht ausreichend. Daher blieb PTS einer der Punkte für den Nachfolgeprozess zum Melker Abkommen („Follow-up“). Diese Tatsache bildete die Grundlage und definierte die Zielrichtung des vorgeschlagenen Projekts.

Das Kernkraftwerk Temelín muss als Spezialfall angesehen werden: Auslegung und Konstruktion wurden in der früheren Sowjetunion durchgeführt, die Herstellung erfolgte zumindest teilweise in der damaligen Tschechoslowakei unter Russischer Aufsicht. Nach der politischen Reorganisation Osteuropas wurde der Bau unter Einbeziehung westlicher Technologie von Westinghouse mit Verantwortlichkeit des Betreibers fertiggestellt. Die Genehmigung erfolgte im legistischen Rahmen der Tschechischen Republik.

Das Projekt PN9 “Reaktordruckbehälterintegrität und Sprödbruchsicherheit unter Thermoschock“ (PTS-Analyse) beschäftigt sich mit dem Versagen des Reaktordruckbehälters (RDB) als Folge einer möglichen Hochdrucktemperaturschock-Transiente. Im Fall der kritischsten Transienten steht der Primärkreislauf unter hohem Druck. Dies ist eine der Hauptsorgen im Rahmen der Reaktorsicherheitsanalyse, da die Rückhaltefunktionen des Reaktordruckbehälters bezüglich Druck und radioaktivem Inventar entsprechend der Auslegung nicht redundant sind. Ein Versagen dieser Komponente würde zu einem katastrophalen Unfall führen.

Überlegungen zur Integrität des Reaktordruckbehälters und ebenso der Ausschluss eines Sprödbruchversagens durch Thermoschock im Kernkraftwerk Temelín sind wesentliche Inhalte seiner Defence-in-Depth Vorkehrungen und daher für Österreich von äußerster Wichtigkeit.

PTS-Ereignisse sollten eine sehr geringe Eintrittsfrequenz haben, da sie signifikante Auswirkungen hinsichtlich des Versagens von mindestens einer gesamten Barriere (Hauptkühlmittelkreislauf) haben können. Da PTS nicht explizit bei der Auslegung vieler älterer, derzeit in Betrieb befindlicher Kernkraftwerke betrachtet wurde, wurden bei den meisten dieser Anlagen erhebliche Anstrengungen unternommen, um Thermoschock unter Druck während des Anlagenbetriebs, aber auch bei Null-Leistung, während eines Abschaltvorgangs und während der Abschaltung zu vermeiden. PTS-Vermeidung wurde als wichtiger Sicherheitsfaktor erkannt und konsequenterweise umfassend und systematisch untersucht.

Bei Anwendung der heute üblichen Sicherheitsphilosophie umfasst die Untersuchung der Reaktordruckbehälter-Integrität und Sprödbruchsicherheit bei Thermoschock, auch Vorkehrungen zur Vermeidung exzessiver Versprödung und anderer negativer Veränderungen der RDB-Werkstoffeigenschaften, sowie weiterer Thermoschock-Konsequenzen.

Das Projekt PN9 besteht aus zwei einander ergänzenden Teilen (horizontal und vertikal), wobei das horizontale Segment eine Bewertung der Prinzipien, Normen und der üblichen Vorgangsweisen anstrebt, während im vertikalen Teil eine rechnerische Analyse von abdeckenden PTS-Referenzstörfallabläufe durchgeführt wird.

In Lichte des umfangreichen Themenkreises von PN9 werden Untersuchungen und Ergebnisse zu all den Fragenstellungen behandelt [ANNEX A], die im Zuge des Monitoring und auch im Rahmen des ExpertInnen-Treffens diskutiert wurden.

Das ExpertInnen-Team konzentrierte sich bei der Durchführung der Auswertung auf die ingenieurmäßige Vorgangsweise, die von ČEZ für die Genehmigungsverfahren für den Reaktordruckbehälter des KKW Temelín durch SÚJB (tschechische Behörde für Nukleare Sicherheit) gewählt wurde.

Beide Teile beziehen sich auf die Zusammenstellung von Informationen zum Versprödungsverhalten der Reaktordruckbehälter(-Stähle) des KKW Temelín, der Vorgeschichte des Druckbehälterwerkstoffs und dessen Ausnutzung, Alterung und Anfälligkeit für Beeinträchtigungen durch Thermoschock.

III. Vorbereitende Arbeiten und durchgeführte Hauptaufgaben im Projekt PN9

Die wesentlichen Aufgaben, die von dem österreichischen ExpertInnen-Team zu erfüllen waren, betrafen zunächst die Durchführung der **Projekt-Milestones (PM)**, wie vom Auftraggeber, dem Umweltbundesamt gefordert. Etliche vorbereitende Arbeiten wurden zur Unterstützung der Hauptaufgaben ausgeführt und diese werden hier ebenfalls angesprochen.

Für die verschiedenen Schritte zu einem Nachweis der strukturellen Integrität des Reaktordruckbehälters wurde der Stand von Wissenschaft und Technik zusammengestellt, um die tschechische Vorgehensweise damit vergleichen zu können. Der Nachweis der strukturellen Integrität eines Reaktordruckbehälters umfasst folgende Punkte:

- Bewertung der Reaktordruckbehälter-Qualität (RDB) bezüglich Auslegung, Konstruktion und Herstellung
- PTS-Analyse
- Bestrahlungsprogramm
- Programm für Zerstörungsfreie Prüfungen
- Modifikationen des Reaktorkerns
- EOPs (Emergency Operation Procedures)

Zur Zeit der Herstellung des Reaktordruckbehälters des Kernkraftwerkes Temelín war die vorliegende Kenntnis über die Versprödung, dass die Verunreinigungen mit Kupfer und Phosphor Ursachen für problematisches Bestrahlungsversprödungsverhalten der WWER Reaktordruckbehälterstähle verursachen. Deswegen wurde der WWER-1000 Reaktordruckbehälterstahl hinsichtlich Kupfer und Phosphor gereinigt. Um bessere Warmbearbeitbarkeit zu erzielen, wurde der Nickelgehalt der Legierung angehoben. Nur wenige Jahre später stellte sich heraus, dass gerade dieser höhere Nickelgehalt eine neue Art von Versprödungsmechanismus herbeigeführt hat. Ein moderner Reaktordruckbehälterwerkstoff, der nach dem Stand der Technik hergestellt wird, wäre ein Stahl, der für geringste Bestrahlungsversprödungsneigung optimiert ist.

Aus diesem Grund ist es fragwürdig, die tschechische Vorgangsweise bei den Maßnahmen für die Reaktordruckgefäßintegrität mit weiteren Genehmigungsanforderungen zu vergleichen, die von der Annahme von optimierten Werkstoffen ausgehen. Die tschechische Genehmigungsgrundlage wird für existierende Kernkraftwerke entwickelt, die nicht nach dem gegenwärtigen Stand von Wissenschaft und Forschung entworfen und gebaut werden. Die grundlegende Forderung nach Reaktordruckbehälterintegrität, nämlich die Verwendung optimierter (nicht zur Versprödung neigende) Stähle, ist mit den Reaktordruckgefäßen in Temelín nicht erfüllt.

Während vorangegangener ExpertInnen-Treffen wurden von den tschechischen Experten im Rahmen des Trialoges und beim Projekt PN2 "Hochenergetische Rohrleitungen auf der 28,8 m Bühne" (AQG/WPNS länderspezifische Empfehlung) [Item No.1]) Informationen zu Folgendem gesammelt:

- Betriebliche Begrenzungsabhängigkeiten für Druck-Temperatur (berechnet in Übereinstimmung mit der Westinghouse Methodik)
- Erste Versuche mit der Druck-Temperaturschock Auswertung in Hinsicht auf den doppelendigen Rohrleitungsabriss (DEGB) einer sekundären Hauptdampfleitung auf der 28,8 m Bühne
- Änderungen zum WWER-1000 Reaktordruckbehälter-Überwachungsprogramms in bezug auf die Ausmerzung der offensichtlichen Mängel solcher Programme
- Werkstoffeigenschaften der Reaktordruckbehälterstähle und der Schweißwerkstoffe im Kernkraftwerk Temelín
- Qualifikationsansatz für ZfP (zerstörungsfreie Prüfungen).

ExpertInnen-Treffen (PM3)

Die vorbereitenden Arbeiten für das ExpertInnen Treffen hatten die Ausarbeitung des Briefing-Materials und eine Briefing-Sitzung für die österreichische Delegation zum Inhalt. Die Grundlagen für Anforderungen an eine PTS-Analyse, der Einrichtung eines Überwachungsprogramms, einschließlich der bekannten tschechischen Modifikationen des Überwachungsprogramms im Vergleich zum ursprünglichen WWER-1000-Überwachungsprogramm, Übereinstimmungen und Unterschiede zum Stand von Wissenschaft und Technik werden beschrieben und in Hinblick auf deren sicherheitstechnische Relevanz kommentiert.

Das ExpertInnen-Treffen, das im Rahmen der „Schlussfolgerungen aus dem Melker Abkommen und Follow-up“ für die erste Hälfte 2004 geplant war, wurde bei SÚJB in Prag am 24./25. Mai 2004 durchgeführt.

Das ExpertInnen -Treffen in Prag 2004 hat sich in Vorbereitung und Durchführung, wie von tschechischer Seite vorgesehen, auf die Durchführung der Hochdruck-Thermoschock Analyse (PTSA) und betriebliche Vorkehrungen zu Hochdruck-Thermoschock Vorgängen konzentriert. Andere Themen mit Bezug auf die Integrität des Reaktordruckbehälters wurden beim ExpertInnen-Treffen nicht behandelt, und zwar die Aushalsungen der Hauptkühlmittelleitungen, die Steuerstabantriebsdurchführungen im Reaktordruckbehälterdeckel, die Durchführungen für die Kerninstrumentierung, die Dichtigkeit des Deckelflansches, sowie wesentliche Einflüsse von Umgebungsbedingungen und anderen Schädigungsmechanismen, die zum Integritätsverlust beitragen, wie die Hauptkühlmittelchemie, Wasserstoffdiffusion, Korrosion, Belastungszyklen, das Verhalten bei Schweren Unfällen, ebenso auch die Integrität erhaltende Maßnahmen und Überwachungsmaßnahmen, welche die Anwendbarkeit des Leck-vor-Bruch Kriteriums sicherstellen, sowie die Messeinrichtungen zur Erfassung von Leckagen. Während des ExpertInnen-Treffens hielt die tschechische Seite 16 Vorträge, die im Rahmen dieses Berichts in den entsprechenden Kapiteln behandelt werden.

Zusätzlich zu den Vorträgen beim ExpertInnen-Treffen hielten die tschechischen Experten eine zweite Vortragsreihe am 7. Oktober 2004 in Řež, um die Information vom ExpertInnen-Treffen zu vervollständigen. Dadurch wurden wichtigen Klarstellungen vermittelt. Diese und die Ergebnisse der Diskussionen, die im Anschluss an die Vorträge geführt wurden, sind in diesen Bericht ebenfalls eingefügt worden.

Erstellung des Preliminary Monitoring Report (PMR) (PM 4)

Die Veröffentlichung der Auswertung der tschechischen Vorträge während des ExpertInnen-Treffens in dem Preliminary Monitoring Report (PMR) im Vergleich mit internationaler Praxis und den tschechischen gesetzlichen Grundlagen war beabsichtigt gewesen. Die Ergebnisse der abdeckenden Referenzstörfall-Berechnungen, die zur Unterstützung des Monitoring durchgeführt wurden, gehen in die Argumentation ein [ANNEX D].

Wegen der zur Verfügung stehenden, begrenzten Zeit und andererseits der Notwendigkeit zusätzliche Informationen vom Oktober-Treffen und dem Bilateralen Treffen 2004 zu berücksichtigen, wurde vom Österreichischen Bundesministerium für Land- und Forstwirtschaft, Umwelt und Wasserwirtschaft schließlich entschieden, die Veröffentlichung des PRM zu übergehen.

Bilateral Meeting (PM5)

Das 13. Bilaterale Treffen gemäß dem Übereinkommen zwischen der Österreichischen Bundesregierung und der Regierung der Tschechischen Republik über Themen von beiderseitigem Interesse auf den Gebieten der Nuklearen Sicherheit und des Strahlenschutzes fand am 29. und 30. November 2004 in Dolni Dunajovive, Tschechische Republik, statt. Bei diesem Anlass wurden die vorläufigen Ergebnisse des Monitoring-Prozesses der tschechischen Delegation vorgestellt und deren Antworten diskutiert. Eine Übersicht über die Aktivitäten, die auf das Bilaterale Treffen folgen werden, wurde geliefert; es wurde ins Auge gefasst zusätzliche Informationen über die mit RPVI und PTS zusammenhängenden Themen in zukünftigen Bilateralen Treffen zu behandeln.

Final Monitoring Report (PM6)

Die Auswertung der zusätzlichen Informationen, die von den tschechischen Experten zur Verfügung gestellt wurden und der weiteren Ergebnisse aus Pilotuntersuchungen, wurden in den vorliegenden Endbericht (Final Monitoring Report) eingefügt.

IV. Hauptergebnisse

IV.1 Reaktordruckbehälter-Integrität und PTS-Analysen

Themen der RDB-Integrität (RPVI), insbesondere in Zusammenhang mit Thermoschockbedingungen wurden in der Tschechischen Republik in bezug auf das KKW Temelín nach Erkenntnissen des österreichischen ExpertInnen-Teams in der folgenden Weise behandelt:

Entsprechend den Normen und Vorschriften praktisch aller Länder muss eine PTS-Analyse vor Inbetriebsetzung des Reaktors als Teil des Genehmigungsantrags zum Nachweis der Sprödbruchsicherheit des Reaktordruckbehälters während der gesamten geplanten Betriebszeit durchgeführt werden.

Das KKW Temelín wurde ohne eine derartige PTS-Analyse in Betrieb genommen. Die Genehmigungsbehörde akzeptierte die betrieblichen Druck-Temperatur-Grenzkurven (Fahrdiagramme), die entsprechend der von Westinghouse erarbeiteten Methode erstellt wurden, als vorläufigen Nachweis der Reaktordruckbehälter-Sprödbruchsicherheit.

Das österreichische ExpertInnen-Team hat die betrieblichen Druck-Temperatur-Grenzkurven (Fahrdiagramme) nach dem Westinghouse-Konzept nicht als angemessenen Ersatz für eine PTS-Analyse angesehen. Zudem wurden die entsprechenden Rechnungen auf nicht-konservativen Annahmen aufgebaut[9]. Daher stellt die „Roadmap" bezüglich der von der tschechischen Seite durchzuführenden Aktionen hinsichtlich RPVI fest, dass ein Schritt-für-Schritt-Verfahren unter voller Berücksichtigung der IAEA-Richtlinien für die PTS-Analyse anzuwenden sei.

Die Präsentationen beim ExpertInnen-Treffen über die ersten Ergebnisse von PTS-Analysen im Rahmen von Projekt PN2 (Conclusions of the Melk Process and Follow-up; Item No. 1: High Energy Pipe Lines at the 28.8 m Level) lieferten erste Informationen über das für das KKW Temelín angewandte PTSA-Konzept.

Die Forderung für die Reaktordruckgefäßintegrität, nur optimierte, d. h. bestrahlungsunempfindliche Stähle zu verwenden, wird bei den Temelín Druckgefäßen nicht erfüllt.

<u>Normvorschriften und Stand von Wissenschaft und Technik</u>

Die Konstruktion des KKW Temelín wurde unter der Aufsicht der ehemaligen Sowjetunion entsprechend der Sowjetischen Normen für Auslegung und Herstellung begonnen. Noch in der späteren Errichtungsphase in der vormaligen Tschechoslowakei und der späteren Tschechischen Republik war das Russische Regelwerk die gültige gesetzliche Genehmigungsgrundlage.

Nach SÚJB (ExpertInnen-Treffen im Mai 2004) beruhen die gegenwärtig geltenden tschechischen Genehmigungsvorgaben bezüglich RPVI und PTSA auf folgenden Grundlagen:

- Teil IV des Association of Mechanical Engineers of the Czech Republic Standards (ASI-Standard): Bestimmung der Restlebensdauer von Komponenten und Rohrleitungen WWER-Kernkraftwerken)
- Die Instruktionen und Empfehlungen für die Lebensdauerbestimmung von WWER-RDBs und Reaktoreinbauten während des KKW-Betriebs [SUJB 1998]
- IAEA-Richtlinien zur PTSA von WWER-Kernkraftwerken [IAEA 1997]
- Die tschechischen Experten haben inzwischen die so genannte VERLIFE Methodik als nicht-verbindliche Richtlinien zum Nachweis der Sprödbruchsicherheit eines Reaktordruckbehälters entwickelt. Diese VERLIFE Methodik wurde Anfang Mai 2004 von SÚJB zugelassen. Das Gesamtkonzept der VERLIFE Methodik ist im Prinzip mit westlicher Praxis vergleichbar.

[9] Der Hauptbericht informiert im Detail ebenso die Informationen in [ATPP 2001].

Prinzipiell kann der Nachweis der Sprödbruchsicherheit eines Reaktordruckbehälters probabilistisch oder deterministisch erfolgen. Die unterschiedlichen nationalen Bestimmungen erfordern keine bestimmte Art der Nachweisdurchführung, dennoch haben viele Genehmigungsbehörden die IAEA-Richtlinien als Grundlage aufgenommen.

In den verschiedenen Nationalen Regelwerken gibt es unterschiedliche Vorschriften zu Sicherheitsfaktoren, die in den Nachweisen verwendet werden müssen, entweder innerhalb der bruchmechanischen Rechnungen (Russisches Regelwerk), oder bezüglich der postulierten Anrissgröße (deutsches Regelwerk) oder Vorschriften, die mit spezifizierten Bedingungen zusammenhängen (französisches Regelwerk)[10].

Die Möglichkeit, von dem sogenannten WPS (warm pre-stressing = warm vorspannen) Effekt Kredit zu nehmen, bewirkt eine Reduzierung des inhärenten Sicherheitsabstandes. Der Effekt wurde als Bestandteil in das neue Russische Regelwerk aufgenommen und wurde in Deutschland angewandt. Er ist allerdings trotz der erlaubten Anwendung nach dem ASME Code in den USA niemals verwendet worden; WPS ist im französischen Regelwerk nicht vorgesehen. Die IAEA Richtlinien (1997) erlauben, WPS in Anspruch zu nehmen[11].

Obzwar die IAEA Richtlinien für PTSA (1997) als Bestandteil der tschechischen Genehmigungsgrundlagen bezeichnet wurden und in den „Conclusions of the Melk Process and Follow-up“ als Grundlage für die durchzuführenden PTS-Analysen zitiert werden, sind anhand der VERLIFE Methodik keine Sicherheitsfaktoren (im Gegensatz zu den IAEA Richtlinien) und die Anwendung von WPS mit dem 90%-Wert des globalen Lastpfadmaximums (in den IAEA-Richtlinien werden 80% empfohlen) vorgesehen. Dies bedeutet eine erhebliche Abminderung der Konservativität, verglichen mit den Empfehlungen der IAEA-Richtlinien für PTSA.

Von ETE veranlasste Aktivitäten bezüglich der PTS-Analysen

Die von NRI Řež für das NPP Temelín durchgeführten PTS-Analysen sind offensichtlich in Übereinstimmung mit den Empfehlungen der IAEA-Richtlinien und dem internationalen Stand von Wissenschaft und Technik bezüglich der Verwendung validierter Computer-Codes wie RELAP5/mod 3.2 für die allgemeine Thermohydraulik, CATHARE und die ingenieurtechnischen Codes REMIX/NEWMIX für die Durchmischungsrechnungen im Ringraum und den Wärmeübergang zur Reaktordruckbehälter-Wand, sowie SYSTUS für die Strukturanalysen und die Bruchmechanik. Für die Berechnung der Spannungsintensitätsfaktoren ist sowohl die Anwendung von FEM-Computer-Codes, als auch die Anwendung analytischer Ansätze basierend auf Gewichtsfunktionen entsprechend dem neuen Russischen Regelwerk erlaubt.

Die Anwendung der genannten Codes wurde von SÚJB freigegeben.

In den folgenden Kommentaren des österreichischen ExpertInnen-Teams zur Auswertung der Präsentationen beim ExpertInnen-Treffen werden auch die Ergebnisse der eigenen Rechnungen für die abdeckenden Referenzstörfälle berücksichtigt. Die Schlussfolgerungen sind unter V.1 zusammengefasst.

Der Sprödbruchsicherheitsnachweis für den Reaktordruckbehälter erfolgt durch die Bestimmung des Sicherheitsabstandes zwischen der maximal erlaubten kritischen Sprödbruchübergangstemperatur T_k^a und dem aktuellen Zustand des Reaktordruckbehälter-Werkstoffs T_k. Die maximal erlaubte kritische Sprödbruchübergangstemperatur ergibt sich aus den bruchmechanischen Berechnungen der Lastpfade für jeden postulierten Anriss in Kombination mit jeder ausgewählten Störfalltransiente.

[10] Der Hauptbericht informiert über Details zu den Regelwerken und zu deren Anwendung.

[11] Die IAEA Richtlinien warnen, dass die Anwendung von WPS sorgfältig bedacht werden müsse, da die Anwendbarkeit bei versprödeten Werkstoffen nicht gesichert ist.

Die Anwendbarkeit des WPS-Effektes wird nach wie vor von der internationalen Fachwelt wegen der theoretischen und experimentellen Unsicherheiten kontrovers diskutiert. Der aktuelle Reaktordruckbehälter-Werkstoffzustand wird durch die Bruchzähigkeitskurve beschrieben, wobei T_k mit Hilfe der Formeln des Russischen Regelwerks, unter Anwendung der dort spezifizierten Versprödungskoeffizienten, ermittelt wird. Die experimentellen Daten aus dem Überwachungsprogramm sollen die Konservativität dieser Koeffizienten bestätigen.

Selbst in der aktuellen Situation mit Szenarien für PTS-Analysen, die nicht alle worst-case Bedingungen berücksichtigen, und der Vernachlässigung von Sicherheitsfaktoren, ergeben sich bei der tschechischen Analyse für den maximal erlaubten Wert der Sprödbruchübergangstemperatur T_k^a Werte, die extrem nahe den EOL (end-of-life)-Werten für T_k von WWER-1000 Werkstoffen liegen. Diese Reaktoren erfahren extreme Thermoschockbelastungen, da eine Kompensierung bei Totalabriss einer Hauptkühlmittelleitung durch die erhöhte Kapazität der Notkühlsysteme möglich ist, wobei großen kalte Wassermengen während der Notkühlmaßnahmen in den Ringraum gedrückt werden. Die Notkühlung führt zu drei Effekten, welche für den Thermoschock wichtig sind:

1. Die inneren Oberflächenschichten der Reaktordruckbehälterwand kommen sehr schnell in den Temperaturbereich 80 ÷ 20 [°C].
2. Die hohe Leistungsfähigkeit der Notkühlsysteme führt zu kalten Zungen im Ringraum und zu steilen Temperaturgradienten über die Druckbehälterwanddicke.
3. Für kleine und mittlere Lecks führt die schnelle Kompensierung des Kühlmittelverlusts durch den hohen Strömungsdurchsatz aus den Notkühlsystemen zu einem frühen und raschen Druckanstieg im Primärkreis, was zur Thermoschockbelastung beiträgt.

Werden international empfohlenen Sicherheitsfaktoren bei der Berechnung der Spannungsintensitätsfaktoren (SIF) für Temelín berücksichtigt, dann werden die kritischen Versprödungsbedingungen erheblich vor Ende der projektierten Betriebszeit (EOL) für das KKW Temelín eintreten. Diese Ergebnisse bekräftigen die von der Sprödbruchsicherheitsanalyse für die WWER-1000-Reaktoren aufgezeigte, ungünstige Situation, die durch die PTSA auch offensichtlich wurden. Nachdem die kritischsten Transienten bei den mittleren Leckgrößen im Primärkreislauf bisher noch nicht gerechnet wurden, kann sich eine noch ungünstigere Situation herausstellen.

Insgesamt ist eine wesentliche Abminderung der Sicherheitsabstände die Folge, wenn das Weglassen der Sicherheitsfaktoren bei der Reaktordruckgefäßauslegung geübte Praxis werden sollte[12].

IV.2 Aktivitäten in Temelín zur Überwachung der Werkstoffversprödung (Bestrahlungsprogramm) und Werkstoffeigenschaften

Überwachungsprogramm

Im Prinzip kann die Werkstoffversprödung durch Neutronenbestrahlung basierend auf den jahrelangen experimentellen Untersuchungen an Reaktordruckbehälter-Stählen vorhergesagt werden, die während der Entwicklung eines Reaktortyps durchgeführt wurden. Die Ergebnisse einer solchen breiten experimentellen Datenbasis wurden jeweils zur Bestimmungen der Werkstoffeigenschaften in den nationalen Regelwerken herangezogen. Daneben wird in praktisch allen Ländern die Werkstoffversprödung der anlagespezifischen Reaktordruckbehälterwerkstoffe (abgesehen von den allerersten Kernkraftwerken) über die gesamte Betriebszeit hinweg mit einem so genannten Bestrahlungsprogramm überwacht.

[12] Diese erstmalige Anwendung könnte das Interesse von WWER-1000 Reaktor-Betreibern wecken, die Anwendung von Sicherheitszuschlägen zu verändern und die IAEA Richtlinien durch wesentliche Verminderung der Sicherheitsfaktoren zu unterlaufen.

Überwachungsprogramme erfordern repräsentative Proben der Reaktordruckbehälter Werkstoffe (repräsentative Grundwerkstoffproben werden aus Abschnitten der Ringe des Reaktordruckbehälters hergestellt, für die Schweißgutproben werden ebenfalls zum Reaktordruckbehälter identische Grundwerkstoffe und Schweißelektroden, sowie identische Herstellungsbedingungen verlangt). Diese Proben werden in Bestrahlungskapseln eingebracht. Die Bestrahlungskapseln des Bestrahlungsprogramms werden im Reaktordruckbehälter so angebracht, dass die Neutronenflussdichte am Ort der Proben gleichmäßig höher als an der Reaktordruckbehälter-Wand im Bereich der aktiven Zone ist, so dass eine beschleunigte (voreilende) Bestrahlungssituation im Vergleich zum Material der Reaktordruckbehälterwand vorliegt, die dann eine Prognose des Versprödungsverhaltens des Werkstoffs erlaubt. Der sogenannte „Voreilfaktor" ist das Verhältnis der Neutronenflussdichten an der Probenposition und an der Reaktordruckbehälter-Wand im Bereich des Reaktorkerns.

Dieses Verfahren liefert eine bestimmte voreilende Bestrahlungsbedingung für jede Probenserie, die prognostizierende Aussagen erlaubt, allerdings eine Begrenzung des erlaubten Voreilfaktors auf maximal 2 erfordert, um eine Verfälschung der Ergebnisse durch hohe Flussdichten zu vermeiden. Die Bestrahlungskapseln werden in regelmäßigen Zeitabständen für die zerstörende Prüfung der Bestrahlungsproben entnommen.

Die ursprünglich für die WWER-1000 Reaktoren vorgesehenen Bestrahlungsprogramme zeigten schwerwiegende Mängel, was durch zwei TACIS-Programme bestätigt wurde:

„Im Rahmen dieser beiden Projekte wurde die Gültigkeit der WWER-1000 Bestrahlungsdaten und anderer experimenteller Ergebnisse auf ihre Repräsentativität hin untersucht. Wegen der geringen Dosiswerte und einer ungenügenden Anzahl von Bestrahlungsproben war die Genauigkeit der Erfassung der Werkstoffversprödung nicht sehr hoch. Es wurde ebenfalls bestätigt, dass die Probentemperatur in den Bestrahlungskapseln vermutlich höher war, als die Temperatur der Reaktordruckbehälter-Wand. In diesem Fall können die Ergebnisse für die Prognose der Reaktordruckbehälter-Versprödung nicht als konservativ angesehen werden."[13] [KRYUKOV 2000]

Die Modifikationen des Bestrahlungsprogramms, die im KKW Temelín eingeführt wurden, eliminieren die evidenten Mängel des ursprünglichen WWER-1000-Bestrahlungsprogramms in Bezug auf Bestrahlungstemperatur, Neutronflussdichte und Neutronendosis an der Probenposition. Die Versprödungsdaten der im KKW Temelín bestrahlten Voreilproben werden daher die ersten zuverlässigen Ergebnisse über die Versprödung von WWER-RDB Werkstoffen liefern.

Materialeigenschaften des Reaktordruckbehälters in Temelín

Die von der tschechischen Seite durchgeführte Bestimmung der kritischen Übergangs-Temperatur T_k wird in Übereinstimmung mit dem Russischen Regelwerk definiert und durchgeführt, die sehr ähnlich der westlichen Vorgangsweise ist. Die Verschiebung dieser Temperatur, die durch die Neutronenversprödung verursacht wird, wird ebenfalls nach Russischen Vorschriften durchgeführt und definiert.

Entsprechend der für das österreichische ExpertInnen-Team zugänglichen Information ist der Ausgangswert (unbestrahlter Zustand) der kritischen Übergangstemperatur T_{k0} in beiden Blöcken für die Schweißnaht Nr.4 am höchsten. Obwohl die Neutronenflussdichte an der Schweißnaht Nr. 4 nur etwa 80% der Neutronenflussdichte an der Schweißnaht Nr.3 ist, muss das Schweißgut der Schweißnaht Nr.4 als führend hinsichtlich der Neutronenversprödung angesehen werden.

[13] Übersetzung durch die AutorInnen

Die prognostizierten Werte für die Neutronenversprödung nach dem Russischen Regelwerk basieren auf experimentellen Ergebnissen aus Testreaktorbestrahlungen des WWER-1000 Stahls 15Ch2MNFAA bei hohen Neutronenflussdichten; es gibt bisher keine Daten mit niedrigen Voreilfaktoren. Die Bestrahlungsexperimente, die am Testreaktor in Řež an Original-Reaktordruckbehälter-Werkstoffen für den Block 1 des KKW Temelín durchgeführt wurden, erfolgten ebenfalls mit hohen Voreilfaktoren (etwa 160 oder höher).

Dies bedeutet, dass mögliche Flussdichte-Effekte (bei gleicher Gesamtneutronendosis ist die Versprödung bei niedriger Neutronenflussdichte höher als bei hoher Neutronenflussdichte) die Ergebnisse beeinflusst haben könnten[14].

Der erste Behälter mit bestrahlten Temelín-Proben wurde im Mai 2004 entnommen; die Auswertung der Proben wird etwa ein Jahr später vorliegen. Bis zu diesem Zeitpunkt wird die PTS-Analyse mit den möglicherweise nicht-konservativen Werten aus dem Russischen Regelwerk durchgeführt. Die VERLIFE Methodik, die vom Betreiber angewandt wird, sieht keine Verwendung von Sicherheitszuschlägen vor, die diese Unsicherheiten abdecken. Das Russische Regelwerk und die IAEA-Richtlinien fordern einen Sicherheitszuschlag von ΔT=10 [K] bezüglich der Unsicherheiten der kritischen Sprödbruchübergangstemperatur. Die U.S. Vorschriften fordern einen Sicherheitszuschlag zur Abdeckung der Unsicherheiten der experimentellen Messungen bei der Bestimmung von RT_{NDT}[15] und den Unsicherheiten der Bestimmung von ΔT_{RTNDT} (15,5 [K] für das Schweißgut und 9,5 [K] für den Grundwerkstoff). Andere Nationale Regelwerke sehen keine Vorschriften für Sicherheitszuschläge zur Abdeckung von Unsicherheiten vor.

Trotz der Tatsache, dass die Daten aus dem Voreilprobenprogramm in Temelín die geforderte Zuverlässigkeit besitzen werden, muss festgestellt werden, dass bei jeder Entnahme eines Bestrahlungsbehälters genau ein Wert zur Abhängigkeit – Versprödung versus Neutronendosis – (Betriebszeit) ermittelt wird. Es herrscht Klarheit darüber, dass das Bestrahlungsprogramm während der projektierten Reaktorbetriebszeit zu keiner statistisch abgesicherten Datenbasis führen kann, die eine zuverlässige Prognose des Versprödungsverhaltens erlauben würde. Da viele Veröffentlichungen darauf hinweisen, dass die Werte der spezifizierten Versprödungskoeffizienten im Russischen Regelwerk als nicht-konservativ angesehen werden müssen, sollten diese Unsicherheiten in der Diskussion der Sicherheitsabstände gegenüber Sprödbruch des Reaktordruckbehälters im Fall von PTS-Störfällen berücksichtigt werden.

Bruchzähigkeitskurven für die PTS-Analyse

Zur Auswertung der Sprödbruchsicherheit oder der Restlebensdauer eines Reaktordruckbehälters mittels PTS-Analyse ist der Vergleich der berechneten Lastpfade für eine ausgewählte Störfall-Transiente mit dem aktuellen Werkstoffzustand des Reaktordruckbehälter-Stahls erforderlich, der sich wegen der zunehmenden Neutronenversprödung verändert. Der Werkstoffzustand wird durch die Bruchzähigkeit als Funktion der Temperatur $K_{Ic}(T-T_k)$ beschrieben.

Die im Rahmen der VERLIFE Methodik definierte Formel für diese Bruchzähigkeitskurve kann als konservativ im Vergleich zu den Formeln im Russischen Regelwerk und der Bruchzähigkeitskurve des ASME Code[16] (die praktisch identisch mit der Kurve des französischen und des deutschen Regelwerks ist) betrachtet werden; sie entspricht der empfohlenen Bruchzähigkeitskurve der IAEA-Richtlinien.

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[14] Die diesbezügliche Frage der Österreichischen ExpertInnen während des Workshops in Prag 2004 über den möglichen Einfluss des Flussdichte-Effekts wurde dahingehend beantwortet, dass dies nicht ausgeschlossen werden könne und Gegenstand zukünftiger Forschungsprojekte sein werde.

[15] RT_{NDT} ist die in westlichen Regelwerken verwendete Referenztemperatur für den Sprödbruchübergang, dieser Wert ist vergleichbar mit dem Tk-Wert des Russischen Regelwerkes.

[16] Unterhalb von etwa 70 [$MPa\sqrt{m}$] ist die ASME Kurve etwas konservativer.

Allerdings konnten die statischen Bruchzähigkeitsmesswerte, die während des ExpertInnen-Treffens 2004 in Prag gezeigt wurden, nicht nachweisen, dass diese Kurve die gemessenen Werte bestrahlter WWER-440-Werkstoffproben konservativ beschreibt. Es gab auch keinen Nachweis, dass alle gemessenen Bruchzähigkeitsdaten von WWER-1000-Werkstoffen konservativ – nach Art einer Master Curve – durch diese Bruchzähigkeitskurve beschrieben werden.

IV.3 Zerstörungsfreie Prüfung (NDT – non-destructive testing)

Reaktordruckbehälter werden regelmäßigen Wiederholungsprüfungen mit zerstörungsfreien Verfahren (ISI: In-Service-Inspections) unterzogen, um etwa vorhandene Defekte aufzuspüren oder zu überwachen. Da die Nachweisverfahren ständig verbessert werden, kann es sein, dass Defekte erstmalig während der Wiederholungsprüfungen detektiert werden – es kann aber auch sein, dass erst durch den Betrieb vorher nicht detektierte Fehler eine detektierbare Größe erreichen.

NDT-Programme müssen durch Verwendung spezieller Prüfkörper, welche für die zu untersuchende Komponente repräsentativ sein müssen, qualifiziert werden.

Der Qualifizierungsvorgang mit dem Testkörper KB190 für die Reaktordruckbehälter-Wand im KKW Temelín wurde offensichtlich erst kürzlich fertig gestellt. Das ist ein Hinweis darauf, dass bisher nur wenige Prüfungen von Wandbereichen in Übereinstimmung mit den qualifizierten und freigegebenen Prüfanweisungen durchgeführt worden sind, die dann qualifizierte Ergebnisse geliefert haben.

Es ist nach wie vor nicht geklärt, ob ein kompletter ZfP Null-Atlas existiert. Die Vergleichbarkeit aller verfügbaren Prüfdaten mit den Ergebnissen qualifizierter Verfahren wurde bisher nicht nachgewiesen.

Der Bereich NDT – zerstörungsfreie Prüfung wird im Detail in einem eigenen Projekt PN10 behandelt: Integrity of Primary Loop Components – Non Destructive Testing (NDT) [Item No. 4][17].

IV.4 Kernauslegung – Neutronendosis

Abschätzungsrechnungen für die Neutronendosis an der Reaktordruckbehälter-Wand sind hinsichtlich der Rechenverfahren sehr empfindlich. Wegen der großen Dämpfung der Neutronenstrahlung zwischen Kern und Reaktordruckbehälter-Wand ist die berechnete Neutronendosis vom verwendeten physikalischen Modell für Kern und Kerneinbauten stark abhängig, sowie von dem mathematischen Modell zum Neutronentransport. Die genaue Bestimmung der Neutronendosis an der Reaktordruckbehälter-Wand ist schwierig, Vergleiche zwischen gemessenen und berechneten Werten zeigen unterschiedliche Übereinstimmung für verschiedene WWER-Reaktortypen und unterschiedliche Kernbeladungsvarianten.

Im Kernkraftwerk Temelín wurde das ursprüngliche Konzepts des russischen Herstellers für den Reaktorkern durch ein neues Konzept von Westinghouse abgelöst. Es ist nicht bekannt, in welchem Umfang dieses Konzept validiert ist. Da es sich um eine Art Prototyp-Anordnung handelt – bis zur Fertigstellung des KKW Temelín wurde keine wesentliche Modifizierung des Reaktorkerns von WWER-1000-Reaktoren durchgeführt – muss angenommen werden, dass ein umfangreicher Validierungsprozess abgeführt worden ist.

[17] The Items are related to ANNEX I of the “Conclusions of the Melk Process and Follow-up”.

IV.5 EOPs

Während des ExpertInnen-Treffens in Prag wurde ein Vortrag darüber gehalten, wie die Temelín Emergency Operation Procedures (EOPs) die Druck-Thermoschock berücksichtigen. Der Stand von Wissenschaft und Technik für prozedurale Aspekte des Thermoschocks erfordert die Verfügbarkeit von EOPs (Emergency Operating Procedures) in der Anlage, um potentielle PTS-Störfälle identifizieren und kontrollieren zu können, sowie um den Reaktor sicher abfahren zu können, ohne Versagen der druckführenden Kühlsysteme und ohne das Auftreten einer nichtadäquaten Kernkühlung. Ziel der EOPs ist die Vermeidung einer Kernschmelze entsprechend dem dritten Niveau von „Defence-in-Depth“. Sollten dennoch Bedingungen auftreten, die zu einer Kernschädigung führen, müssen so genannte SAMGs (Severe Accident Management Guidelines) verfügbar sein, um die Schädigung des Reaktorkerns begrenzen zu können und die Konsequenzen einer Kernschmelze zu mildern. Symptomorientierte Störfallprozeduren (EOPs), ein Ergebnis eines CEZ-Westinghouse Projekts wurden 1998 in Temelín eingeführt. In ähnlicher Weise sind die SAMGs im Stadium der Einführung, das mit Ende 2004 abgeschlossen sein sollte. CEZ hat einen standardisierten und anerkannten Ansatz für die Prozeduren gewählt, die mit den Thermoschock-Ereignissen umzugehen erlauben sollen indem die Westinghouse EOPs und SAMGs eingeführt werden.

Dieser Fragenkomplex wurde detailliert im Rahmen des Projekts PN7 Severe Accidents related Issues – [Item No.7b] behandelt.

IV.6 Trainingsprogramme – QA Programme

Die Implementierung eines ausgedehnten Programms, wie RPVI-Gewährleistung und VERLIFE, bezieht viele Fachbereiche und TSOs ein. Darum müssen ausgedehnte Programme für Training- und Qualitätssicherung vor und während des Durchführungszeitraums erstellt werden, d. h. in diesem Fall während der gesamten Betriebszeit der Anlage.

Die Information zu Trainingsprogrammen für das Personal in Zusammenhang mit PTS-Störfällen ergab ein zufrieden stellendes Bild. Die Maßnahmen sind im Einklang mit Europäischen Vergleichssituationen.

Die Darstellungen hinsichtlich des QA-Programms betrafen nur einige allgemeine Informationen zu QA-Verfahren und Freigaberichtlinien die von SÚJB auferlegt, und für von NRI durchgeführte Rechneranalysen zur Anwendung kamen.

IV.7 Die Position von SÚJB

Die Erwartungen aus der Positionierung von SÚJB wurden während des ExpertInnen-Treffens weitgehend geklärt, wobei allerdings die Substanz der Entscheidungsgrundlagen z. B. hinsichtlich der Genehmigung des VERLIFE-Konzeptes, ebenso wie der Zeitplan für die Fertigstellung der PTS-Analysen nicht diskutiert wurden.

Beim ExpertInnen-Treffen wurden alle entscheidenden Fragen in einer sehr allgemeinen Art angesprochen, spezifische Informationen wurden nur bei bestimmten Fragestellungen erhalten. Das ExpertInnen-Team erhielt Einblick in wesentliche Punkte betreffend externe Unterstützung und Unabhängigkeit, die auch in den NRA-Grundlagen angesprochen werden.

Die IAEA IRRT-Mission in der Tschechischen Republik [IRRT 2000, paragraph 1.7.1] hat – neben anderen – eine Empfehlung bezüglich externer und unabhängiger Expertise abgegeben. Es wurde auch festgestellt, dass die Personalstärke die Möglichkeiten der SÚJB mit allen angebrachten Mitteln erweitert werden müssten.

Sowohl im Allgemeinen als auch im Besonderen sollte dies auch im Hinblick auf die Integrität des Reaktordruckbehälters und die PTS-Fragestellungen im Kernkraftwerk Temelín in ausreichendem und effizientem Maß der Fall sein.

V. Schlussfolgerungen

Die Demonstration der Integrität des Reaktordruckbehälters (Reactor Pressure Vessel Integrity, RPVI) während seiner Lebensdauer wird von den tschechischen Experten mittels der VERLIFE-Methodik durchgeführt. Im Vergleich mit dem Russischen Regelwerk und den IAEA-Richtlinien hat das österreichische ExpertInnen-Team herausgefunden, dass die VERLIFE-Methodik, wie sie in Temelín angewendet wird, die Sicherheitsreserven reduziert (d. h. Reduzierung der postulierten Rissgröße, Eliminieren/Reduzierung von Sicherheitsfaktoren, nicht-konservative bruchmechanische Annahmen bei den bruchmechanischen Analysen). In Kombination mit anderen Unsicherheiten, welche die Werkstoff/Versprödungs eigenschaften und offensichtliche Reduzierung der Konservativität in mehrfacher Hinsicht, wird die resultierende Gesamt-Sicherheitsreserve bei RPVI für die Reaktordruckgefäße in Temelín vom österreichischen ExpertInnen-Team als nicht ausreichend angesehen.

Die vollständigen VERLIFE-Methodik Anforderungen und deren Anwendung auf das KKW Temelín waren für das österreichische ExpertInnen-Team nicht verfügbar. Das österreichische ExpertInnen-Team war daher ausschließlich auf die beim ExpertInnen-Treffen vermittelte Information angewiesen.

Das österreichische ExpertInnen-Team stellte weiterhin fest, dass die derzeitige tschechische Vorgehensweise – so wie sie präsentiert wurde – bei der Analyse von Thermoschockbelastung unter Druck (Pressurized Thermal Shock, PTS) im Hinblick auf das Konzept, die Methodik und die angewandten Computer-Codes, mit dem Stand der Technik übereinstimmt. Die schwersten Störfallabläufe, die analysiert wurden, sind gut vergleichbar mit jenen, die nach heutigem Wissensstand für WWER-1000 Anlagen als repräsentativ angesehen werden. Alle Gruppen von Unfällen, die für eine PTS-Analyse wichtig sind, wurden betrachtet.

Andrerseits bleibt eine Anzahl von Punkten erklärungsbedürftig:

- Die Grundlage für die Analysen erscheint unzureichend. Obgleich alle Gruppen von Unfällen, die für eine PTS-Analyse wichtig sind, untersucht wurden, hat der Zeitrahmen der Simulation in manchen Fällen möglicherweise die kritischen Belastungen des Reaktordruckbehälters nicht erfasst, da die Simulationen möglicherweise gerade beendet wurden, bevor ein erneuter Druckanstieg stattgefunden hätte. Innerhalb einiger Unfall-Gruppen können die betrachteten Abläufe in manchen Fällen nicht als die kritischsten angesehen werden. Für einige Transienten ist es erforderlich, dass die Notfall-Betriebsmaßnahmen innerhalb eines schmalen Zeitfensters ausgeführt werden, um Sprödbruch des RDB zu vermeiden.
- Es bestehen offensichtliche Reduktionen der Konservativität. Einzelne VERLIFE-Kriterien sind schwächer als die Erfordernisse der IAEA-Richtlinien. Unter Anwendung der Werte für postulierte Rissgröße, Sicherheitsfaktoren und WPS (Warm Prestress Effect)-Kriterium, wie sie den IAEA-Richtlinien entsprechen, würde es keinen erfolgreichen Nachweis der Erfüllung der geforderten RPVI über die gesamte Betriebszeit geben.
- Die identifizierten Unsicherheiten bezüglich der PTS-Analysen betreffen beispielsweise: die thermo-hydraulische Modellierung von Unfallabläufen, die Modellierung des Mischungsverhaltens, das Versprödungsverhalten, sowie auch die Ausgangswerte der Sprödbruchübergangstemperatur der RDB-Werkstoffe, die Bestimmung von Fluenzen und die Einführung von Maßnahmen zur Fluenzminimierung, sowie Bereiche bei den Wiederkehrenden Prüfungen (ISI), für welche die Qualifikation bisher nicht beendet wurde. Das sind weitere kritische Bereiche, die einer Klärung bedürfen.
- Die Konservativität wird zusätzlich durch die Einbeziehung einer intakten Plattierungs-Zone als Strukturverstärkung in das Finite-Element-Modell, einschließlich nicht-konservativer Annahmen bei den bruchmechanischen Ansätzen an der Grenzfläche Plattierung/ ferritischer Stahl reduziert (wie bei einer Pilot-Untersuchung des österreichischen ExpertInnen-Teams bestätigt wurde). Es wurden nicht alle Typen von Unterplattierungsrissen untersucht.

Bezüglich des Voreilprobenprogramms, das den Fortschritt der Versprödung verfolgt, insbesondere hinsichtlich der Positionierung der Proben, kann festgestellt werden, das eine erhebliche Verbesserung im Vergleich zu anderen WWER-1000-Reaktoren derselben Generation erzielt wurde.

Folgerichtig sollte der zukünftige Austausch von Informationen über die Reaktordruckgefäßintegrität und den Thermoschock vor allem folgende Themen abdecken:

- Im Hinblick auf PTS-Analysen sind die Konsequenzen zusätzlicher kritischer Bedingungen, sowie eines erweiterten Zeitrahmens für manche der berechneten Abläufe von Interesse; ebenso die Betrachtung aller Rissgrößen und Risspositionen, welche für die Bruchmechanik relevant sind (einschl. von Stabilitätsbetrachtungen).
- Das Fortschreiten der Versprödung und die Gegenmaßnahmen, die ergriffen wurden, sollten weiter beobachtet werden. Dies schließt Ergebnisse von Bestrahlungsproben für beide Blöcke des KKW Temelín ein, insbesondere die Ergebnisse bestimmter Proben mit erhöhter kritischer Ausgangssprödbruchübergangstemperatur, die in Block 2 bestrahlt werden.
- Der Vergleich der Werkstoffeigenschaften, die bei den Qualifikations-Tests, den erweiterten Abnahme-Tests sowie dem Lebensdauer-Auswertungsprogramm bestimmt wurden, mit den Daten aus dem Bestrahlungsprogramm ist von Interesse, um die Streuung der Materialeigenschaften bewerten zu können.
- Maßnahmen zur Abschwächung des Versprödungsfortschrittes, insbesondere Änderungen beim Nachbeladen mit Brennstoff und Anreicherungsabänderungen, sind von Interesse.

Im Laufe eines weiteren Informationsaustausches könnten die Themen, die hier aufgelistet wurden, mit Themenbereichen kombiniert werden, die bei Punkt 4 (Zerstörungsfreie Prüfung) für zukünftigen Informationsaustausch verbleiben – nämlich des zuverlässigen Nachweises aller PTS-relevanten Defekte.

Schlussfolgerungen im Detail

Bei der Gegenüberstellung der Präsentationen aus Anlass des ExpertInnen-Treffens und den anerkannten internationalen Richtlinien, Empfehlungen und Entscheidungen gelangte das österreichischen ExpertInnen-Team zu folgenden Erkenntnissen: Eine Vielzahl dieser Erkenntnisse haben auch das Ergebnis von generischen Berechnungen und Untersuchungen zur Grundlage, die während der Vorbereitung des ExpertInnen Treffen durchgeführt worden sind.

- Die österreichischen ExpertInnen begrüßen, dass die tschechische Seite nicht mehr die betrieblichen Druck-Temperatur-Grenzkurven als angemessenen Nachweis für die Vermeidung nicht-akzeptabler PTS-Transienten ansieht.
- Das globale RPVI-Konzept hinsichtlich der Durchführung der PTS-Analysen entspricht offenbar dem Stand von Wissenschaft und Technik bzw. Praxis und den IAEA-Richtlinien. Mit der Art und Weise, wie die Sicherheitsvorkehrungen aus IAEA-Richtlinien in der VERLIFE Methodik interpretiert werden, werden sie nicht unerheblich vermindert.
- Das vorgestellte tschechische Konzept für die PTS-Analyse (Teil von VERLIFE) kann hinsichtlich Konzeptbasis, Methodik und angewandte Computer-Codes als mit dem Stand von Wissenschaft und Technik übereinstimmend angesehen werden.
- Offensichtlich wurden alle thermohydraulischen Berechnungen mit Computer-Codes durchgeführt, die für die Anwendung für WWER-1000 Reaktoren validiert worden sind. Nach der Fertigstellung dieser RPVI/PTS-bezogenen TH-Analysen können diese als vollständig und solide angesehen werden. TH-Analysen sollten eine verlässliche Grundlage für die Auswahl der Störfall-Kandidaten hinsichtlich der nachfolgenden Durchmischungs- und Wärmeübergangs-Berechnungen sein. Die Verwendung von Annahmen, welche für den jeweiligen Zweck nicht konservativ sind und aus diesem Grund Auswirkungen auf die Sicherheit haben, sollte überprüft werden.

- Die schärfsten Transienten sind sicher mit denjenigen vergleichbar, die –entsprechend gegenwärtigem Wissensstand – für WWER-1000-Anlagen als repräsentativ angesehen werden. In einigen Bereichen könnte es sein, dass der in der Simulation betrachtete Zeitabschnitt den zu einer kritischen Belastung des RDB führenden Zeitpunkt in der Simulation nicht erfasst hat, da der der Wiederanstieg des (System)drucks eben noch nicht so frühzeitig (d. h. vor Abbruch der Simulation) erfolgt.

Bezüglich der Punkte „Durchmischung und Wärmeübergang“:

- Die Durchmischungsberechnungen für die Störfalltransienten in den PTS-Analysen sind offenbar in Übereinstimmung mit dem Stand von Wissenschaft und Technik und der internationalen Praxis, sowie vergleichbar zu den Berechnungen für andere WWER-1000-Reaktoren.

Bezüglich FEM-Berechnungen und Bruchmechanik:

- Die angewandten Computer-Codes für die FEM-Simulation und die Berücksichtigung des elastoplastischen Materialverhaltens sind als in Übereinstimmung mit dem Stand von Wissenschaft und Technik anzusehen. Sobald die Analysen mit Hilfe dieser Methodik fertiggestellt sein werden, wird die erhaltene PTS-Analyse als vollständigste bisher für WWER-1000 durchgeführte PTS-Analyse überhaupt anzusehen sein.
- Die IAEA-Richtlinien erlauben die Verwendung von postulierten Risstiefen kleiner als die üblicherweise erforderliche ¼ Wanddicke (das sind beim WWER-1000 etwa 50 [mm]) für den Fall, dass die NDT (zerstörungsfreie Prüfungs)-Programme einen sicheren Nachweis entsprechend kleiner Risse erlauben. Für diesen Fall fordern die IAEA-Richtlinien die verpflichtende Anwendung von Sicherheitsfaktoren: Sicherheitsfaktor 2 für die Risstiefe oder Sicherheitsfaktor $\sqrt{2}$ für die Spannung und $\Delta T = 10$ [K] für die durch Neutronenversprödung verursachte Verschiebung der kritischen Sprödbruchübergangstemperatur. In Übereinstimmung mit VERLIFE [PISTORA 2004a] verwenden die tschechischen Experten eine postulierte Risstiefe von 20 [mm] (etwa $^{1}/_{10}$ der Wanddicke also erheblich geringer als ¼ Wanddicke), und setzen keine Sicherheitsfaktoren an (z. B. im Gegensatz zu den IAEA Richtlinien).
- Der tschechische Ansatz weist Abweichungen von den IAEA-Richtlinien [IAEA 1997] betreffend die fehlende Variation von Risstiefe und Rissgeometrie auf. Folgende Untersuchungen wurden nicht dargestellt:
  - die Analyse von sehr kleinen Rissen (a < 6 [mm]),
  - die Analyse großer Risse (a = 20 [mm] bis zu ¼ der Wanddicke), und
  - die Variation des Achsenverhältnisses bis a:c = 1:10.
- Die Vorgangsweise beim Einbeziehen der Plattierung in die FE-Modellierung führt zu einer weiteren Reduktion der Konservativität, nicht nur durch die Nichtberücksichtigung von elliptischen Unterplattierungsrissen, sondern auch durch die Annahme, dass der Spannungsintensitätsfaktor (SIF) an der Grenzfläche Plattierung /Grundwerkstoff zu SIF = 0 verflacht, was nicht der Realität entspricht. Dies wurde durch eigene Rechnungen bestätigt, die im Rahmen des Projekts durchgeführt wurden.
- Das Finite-Elemente-Modell stellt eine Hälfte des Reaktordruckbehälters dar. Dieses Verfahren umfasst keine Berücksichtigung der Überlagerung kalter Zungen, der dehnungsbedingten Verformung des Zylinders, sowie der Wechselwirkung mit dem Druckbehälterboden und dem Deckel (Verformungsbehinderungen). Diese Vorgangsweise ist in Übereinstimmung mit der internationalen Praxis. Die Simulation mit einer Vernetzung des gesamten Druckbehälters würde eine erhebliche zusätzliche Anstrengung bedeuten.

Bezüglich PTSA:

- Alle Störfallgruppen die für eine Behandlung in der PTSA wichtig sind, wurden analysiert. Für WWER-1000 Reaktoren ist das die erste PTSA in einer Vollständigkeit, wie sie bisher noch nicht erreicht worden ist.

- Die Thermoschockbelastungen in WWER-1000-Reaktoren sind extrem hoch. Für postulierte Risstiefen von nur 20 [mm] liegen die T_k^a-Werte in vier Fällen und drei Störfallgruppen unter 70 [°C], bei anderen Reaktortypen findet man keine vergleichbaren Verhältnisse z. B. bei WWER-440-Reaktoren. Dies ist eine Folge der hohen Leistungsfähigkeit der Notkühleinspeisesysteme, die in der Lage sind, große Lecks bis NW 850 zu kompensieren, sie übertragen aber gleichzeitig eine extreme Thermoschockbelastung auf die Reaktordruckbehälterwand.
- Die niedrigsten T_k^a-Werte wurden für kleine bis mittlere Leckgrößen festgestellt, wo zusätzlich zu den Thermoschockbelastungen noch ein vollständiger oder teilweiser Wiederanstieg des Druckes im Primärkühlmittelkreislauf zu verzeichnen sein kann.
- Der Reaktorfahrer muss die genau angepassten Handlungsabläufe (EOPs) zum richtigen Zeitpunkt ausführen, um mit einer Anzahl von Störfalltransienten (PSV41) zu Rande zu kommen und gleichzeitig das Sprödbruchversagen des Reaktordruckbehälters vermeiden. Jedenfalls ist es international unüblich derartige „garantierte" Handlungsabläufe vom Bedienungspersonal zu verlangen, deswegen muss das als wesentlicher Abbau der Konservativitäten bei der Behandlung von Störfällen angesehen werden.
- Einige Störfälle (PSV43) wurden nicht bis zu dem Punkt gerechnet, wo das 90% WPS-Kriterium anwendbar ist.
- In einigen Fällen kann die Definition der Störfalltransienten nicht als die kritischste abgesehen werden: In der Störfallgruppe PRZ SV wurde der Ausfall der Fremdversorgung nicht mit einbezogen, obwohl die Berücksichtigung in den IAEA-Richtlinien gefordert wird. Durch den Ausfall der Fremdversorgung könnte es zu einem höheren Druckanstieg im Primärkreislauf nach dem Wiederschließen des Druckhaltersicherheitsventils kommen.

Bezüglich der von den IAEA-Richtlinien geforderten Sicherheitsfaktoren muss festgestellt werden:

- Die VERLIFE-Methode, die von den tschechischen Experten für das KKW Temelín angewendet wird, verwendet nur postulierte Risstiefen von 20 [mm] (entspricht $^1/_{10}$ der Wanddicke, was erheblich weniger als ¼ der Wanddicke ist), und keine Sicherheitsfaktoren entgegen den IAEA Richtlinien.
- Das von den tschechischen Experten für Temelín angewandte VERLIFE-Konzept verwendet bei der Inanspruchnahme des WPS-Effektes das 90% -Kriterium, obwohl die IAEA-Richtlinien das 80%-Kriterium empfehlen, wenn überhaupt WPS angenommen wird. Diese Modifikation führt zusätzlich zu einer erheblichen Reduzierung der Konservativität, was dem Erfordernis, die Unsicherheiten der Vorhersage der Versprödung zu kompensieren, widerspricht.
- Obwohl die Anwendbarkeit des WPS-Effektes in der internationalen Fachwelt wegen der theoretischen und experimentellen Unsicherheiten noch immer umstritten ist[18], wird der WPS-Effekt für den Sprödbruchsicherheitsnachweis des Temelín-RDBs angewandt.
- Die konsequente Anwendung der IAEA-Richtlinien würde zu einem anderen Ergebnis der PTS-Analyse für das KKW Temelín, als vom Betreiber vorgestellt, z. B. in Fällen, in denen das 80% WPS-Kriterium zusammen mit dem Sicherheitsfaktor $\sqrt{2}$ und $\Delta T = 10$ [K]) anzuwenden wäre.

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[18] Dies betrifft insbesondere die reale Situation in der Komponente und die Temperatur/Druck-Verläufe während eines realistischen Thermoschock-Störfalls.

Zum Überwachungsprogramm im KKW Temelín und zum Versprödungsverhalten der Werkstoffe der ETE-RDBs:

- Eine Grundvoraussetzung für RPVI nach dem Stand von Wissenschaft und Technik, nämlich die Verwendung optimierter, strahlungsunempfindlicher Stähle, ist für die Reaktordruckbehälter in Temelín nicht erfüllt.
- Das modifizierte Bestrahlungsprogramm im KKW Temelín erlaubt die Bestimmung zuverlässiger Versprödungsdaten hinsichtlich der Bestrahlungstemperatur und der Neutronenflussdichte/-dosis an der Position der bestrahlten Proben.
- Das modifizierte Bestrahlungsprogramm führt dazu, dass einige Bereiche des RDB im Bereich der Bestrahlungskontainer in der Nähe der Schweißnaht 4 für die zerstörungsfreie Prüfung unzugänglich sind.
- Die Auswertung der veröffentlichten Ergebnisse von Bestrahlungsproben von WWER-1000 Werkstoffen zeigt unter Berücksichtigung der geschätzten Bestrahlungstemperatur beachtliche Unsicherheiten hinsichtlich der Neutronenversprödung des WWER-1000 Stahls. Aus Versuchsergebnissen hat man erkannt, dass die Spezifikation des Russischen Regelwerks (A_F = 20 für das Schweißgut, 23 für den Grundwerkstoff) nicht als konservativ angesehen werden kann.
- Obwohl die ersten zuverlässigen Ergebnisse für einen WWER-1000 aus dem Bestrahlungsprogramm des KKW Temelín zu Verfügung stehen werden, bleiben die Unsicherheiten über das Versprödungsverhalten des WWER-1000-Stahls erhalten: das RDB-spezifische Bestrahlungsprogramm kann nämlich keine zuverlässige statistische Grundlage für die Voraussage der Werkstoffversprödung liefern, da jeder Satz von Bestrahlungsproben, die entnommen und ausgewertet werden, nur einen einzigen Datenpunkt zu der Korrelation Versprödung–Bestrahlungszeit hinzufügt.
- Die bisher für Temelín-spezifische Werkstoffe ermittelten Versprödungskoeffizienten basieren auf Testreaktorbestrahlung mit hohen Voreilfaktoren. Der Flussdichte-Effekt könnte daher die Bestimmung der Versprödungsfaktoren gegenläufig beeinflusst haben, d. h. die Versprödung könnte in der Realität größer sein, als gemessen.
- Die in den Reaktorpässen ausgewiesenen Werkstoffeigenschaften zeigen, dass die Ausgangswerte der Sprödbruchübergangstemperatur T_{k0} um einige -zig Grad von einer Schweißelektrodencharge zur nächsten schwanken können. Es war nicht möglich, zu überprüfen, ob bei der Bestimmung von T_{k0} die in der VERLIFE-Methodik vorgeschriebene Anwendung des Temperatur-Sicherheitsabstandes δT_M (10 [K] für den Grundwerkstoff und 16 [K] für das Schweißgut), der die Streuung der mechanischen Eigenschaften abdecken soll, bei der angegebenen Auswertung von T_{k0} berücksichtigt wurde.
- Diese Tatsache und die Unsicherheiten hinsichtlich der spezifizierten Versprödungskoeffizienten machen die Berücksichtigung zumindest des von den IAEA-Richtlinien [IAEA 1997] vorgeschriebenen Sicherheitsfaktors ΔT erforderlich.
- Die Schweißnaht Nr. 4 in ETE-1 wurde mit zwei verschiedenen Elektrodenchargen (Sv12Ch2N2MAA, Chargennummer 17084 und 170007) geschweißt; für beide Chargen wurden Bestrahlungsproben hergestellt; das Bestrahlungsprogramm von ETE-1 verwendet aber nur Proben, die mit derselben Schweißelektrodencharge wie Schweißnaht Nr. 3 hergestellt wurden (T_{k0} = –50 [°C]). Das andere Schweißgut mit T_{k0} = –30 [°C] wird erst im Bestrahlungsprogramm von ETE-2 enthalten sein. Aus Sicht der österreichischen ExpertInnen ist dies ein erhebliches Defizit, da die Bestrahlungsergebnisse für dieses Schweißgut mit den für ETE-1 höchsten Ausgangswerten erst mit erheblicher Verzögerung zur Verfügung stehen werden.
- Die in der VERLIFE-Methodologie verwendete Bruchzähigkeitskurve kann als konservativ im Vergleich zu den Bruchzähigkeitskurven anderer nationaler Regelwerke angesehen werden.

Hinsichtlich des NDE/ISI-Programms im KKW Temelín:

- ISI mit ZfP-Ultraschallverfahren für den zylindrischen RDB-Teil kann als erfolgreich qualifiziert angesehen werden. Die Methoden erlauben grundsätzlich den Nachweis aller Arten rissähnlicher Defekte, die als wesentlich im Zusammenhang mit Thermoschock-Störfällen und deren Analyse gelten, wie z. B. Risse nahe der Plattierungsgrenzschicht zum Grundwerkstoff mit einem Halbachsenverhältnis $^{a}/_{c}$ von z. B. 0,3 und unterschiedlichen Tiefenausdehnungen in Abhängigkeit von den PTSA-Risspostulaten. Ein halbelliptischer Riss, der an der Plattierungsgrenze beginnt und 8 [mm] tief in das ferritische Material reicht, scheint für NDT der kritischste Fall zu sein. Obwohl die Qualifizierung mit einem RDB-Testblock die grundlegende Eignung zum Nachweis solcher Defekte mit den angewandten US-Verfahren gezeigt hat, bleiben dennoch Probleme, die noch nicht endgültig gelöst sind.
- Der Testblock enthält die Plattierung nicht unter Bedingung wie an der Schweißnaht und deren Umgebung, wo man ein erhebliches Rauschniveau annehmen muss und daher eine höhere Rate an Falschanzeigen zu erwarten ist, wie auch im Qualifizierungsbericht erwähnt wird. Dies erfordert Gegenmaßnahmen, wie z. B. Wirbelstromprüfung (ECT) in Bereichen mit einer erhöhten Anzahl an US-Anzeigen. Dies ist insbesondere erforderlich, wegen der VERLIFE-Voraussetzung einer intakten Plattierung, speziell in Bereichen von Rissen nahe der Plattierung im ferritischen Wandbereich. Das restliche Ligament zwischen der Rissspitze und der inneren Oberfläche kann nur mit qualifizieren ECT-Methoden geprüft werden. Die Sicherheitsbeurteilung bezüglich des Nichtvorhandenseins von Anrissen, die für Thermoschock von Bedeutung sind, ist zurzeit noch nicht abgeschlossen, weil weder die Qualifikation der Wirbelstromprüfmethode, noch der erforderliche Prüfvorgang mit ihr bis dato abgewickelt worden sind.
- Zwei weitere ISI-Bereiche mit speziellen PTS-Bezügen sind die inneren Bereiche der Eintritts- und Austrittsstutzen und der Anschlussschweißnähte an den Verbindungsstellen des Primärkreislaufes mit dem RDB. Für beide Bereiche wurden Qualifikationen angekündigt, wurden aber nicht fertiggestellt und auch nicht vorgestellt. Von besonderem Interesse sind PTS-relevante Rissgrößen in den Stutzenkanten und in den Verbindungsnähten, um die Schwierigkeiten der NDT-Verfahren hinsichtlich der erforderlichen Auffindbarkeit von Defekten und einer vernünftigen Fehlanzeigenrate beurteilen zu können.
- Unter Berücksichtigung der noch nicht durchgeführten NDT-Arbeiten, die aber für den Nachweis der sicheren Auffindbarkeit PTS-relevanter Risse erforderlich sind, muss festgestellt werden, dass die bisher durchgeführten Inspektionen nur einen Teil der ISI-Erfordernisse erfüllen. Entsprechend den während des ExpertInnen-Treffens erhaltenen Informationen ist die Vervollständigung der PTS-relevanten ISI-Aktivitäten in Vorbereitung, wobei einige Qualifizierungsteile in Arbeit sind, die aber sicher nicht vor der nächsten RDB-ISI fertig sein werden.

Schlussfolgerungen betreffend Kernauslegung und Versprödungsabminderung:

- Die OUT-IN-Strategie ist eine wohlbekannte, überkommene Methode, die Versprödung zu mindern; die ETE-spezifische Information in der Präsentation während des ExpertInnen-Treffens konnte die Frage nicht beantworten, ob eine bestrahlungsvermindernde Modifikation durchgeführt wurde, oder es sich nur um einen Nebeneffekt der Leistungsoptimierung handelt. Die PTS-relevanten Effekte der Dosisreduktion am RDB können nur aus der Fluenz-Verteilung abgeleitet werden. Trotzdem waren die zur Verfügung stehenden Informationen Zeichnungen der Leistungsverteilung.
- Die Feststellung im Rahmen des ExpertInnen-Treffen, der tatsächliche Betrieb würde beachtlich unterhalb der berechneten Fluenzwerte stattfinden, belegt von sich aus nicht, dass die Versprödungsminderung angemessen erfolgt. Das Konzept des Dosisreduktionsmanagements für das Reaktordruckgefäß ist ein wesentliches Element, das während des KKW-Betriebs vorangetrieben werden muss.

- Die Westinghouse Kernauslegung, die in einem WWER-1000 Reaktor verwendet wird, ist die erste ihrer Art, die zu validieren sein wird. Offensichtlich wurde die Anordnung des Kerns bisher aber nicht mit dem Ziel einer Dosisminimierung an der RDB-Wand zur Reduktion der Neutronenversprödung des Stahles verändert. Diese Verbesserung der Eigenschaften des Reaktorkerns soll während einer der nächsten Nachbeladungsabschaltungen stattfinden. Zum jetzigen Zeitpunkt wurden keine genaueren Informationen über die geplanten Änderungen vorgelegt.

Schlussfolgerungen den Übergang von EOPs und zu SAMGs betreffend:

- Umfangreiches Feedback aus den Anlagenanalysen wurde dazu genutzt, die Grundzüge der EOPs und deren Elemente zu einem up-to-date Notstands-Management Hilfsmittel zu machen. Von der Übersichtsdarstellung kann man ableiten, dass das Konzept sich für zielführende Anpassungen eignet. Diesbezügliche Arbeiten sind offensichtlich ein erfolgreicher, laufender Prozess.
- Die EOPs ebenso wie die SAMGs und entsprechende Vorkehrungen wurden in Übereinstimmung mit dem Stand von Wissenschaft und Technik erstellt, unter der Voraussetzung, dass die zu verwendenden Geräte qualifiziert wurden oder auch für die vorgesehenen Einsätze im entsprechenden Betriebsregime qualifiziert wurden.

Schlussfolgerung zu den Punkten Qualitätssicherung und Training:

- Wegen fehlender detaillierter Informationen war es nicht möglich, die Effizienz der Qualitätssicherungsprogramme in Zusammenhang mit den RPVI-Aktivitäten im KKW Temelín zu bewerten. Jedenfalls werden die erzielten QA-Verbesserungen in der Bewertung der Qualitätssicherung gewürdigt.
- Die Sicherstellung und Konsolidierung fundierten Verständnisses des gegenwärtigen Zustandes des RDB und der Anlagensysteme erfordert, dass Betriebsanweisungen und Betriebsführungsstrukturen eingerichtet werden. Die Betriebsführung sollte für einen Ablauf installiert werden, der für die gesamte Lebensdauer der Anlage ausgelegt ist. Die dazu erforderlichen Voraussetzungen sind in angemessenen Dimensionen eingerichtet worden.
- Die Ausbildungs- und Umsetzungsvorkehrungen sind umfassend und mit den Maßnahmen in anderen Kernkraftwerken in Europa vergleichbar. In einzelnen Fällen ist höchst wahrscheinlich der Gesamtlösung gegenüber einer zeitgerechten Einführung der Vorzug gegeben worden.

Schlussfolgerungen zur Position von SÚJB:

- Die Position von SÚJB zu den „PTS-Anforderungen“ gegenüber dem Genehmigungsnehmer ist ein Hinweis auf deren Beobachterfunktion in Hinsicht auf die Sicherstellung der RDB-Integrität und die Einhaltung der PTS-Vermeidung.
- In Übereinstimmung mit der Empfehlung der IAEA IRRT Mission zieht das österreichische ExpertInnen-Team in Erwägung, dass ein erstrebenswertes Ziel die „Stärkung“ von SÚJB ist. Deren personelle Kapazität und Möglichkeiten sollen – der Aussage folgend – mit allen erforderlichen und zu Gebote stehenden Mitteln in geeignetem Maße erhöht werden, hier insbesondere in RPVI- und PTS- Belangen.

VI. PTS – Empfehlungen für ein weiterführendes Monitoring hinsichtlich RPVI/PTS

Das österreichische ExpertInnen-Team empfiehlt folgende Punkte mit hoher Vordringlichkeit im Rahmen des laufenden bilateralen Übereinkommens zwischen der Bundesrepublik Österreich und der Tschechischen Republik weiterzuverfolgen. Dies betrifft die Implementierung und Ergebnisse des RPVI-Programms, VERLIFE und PTSA. Zusätzlich, weil der laufende RPVI/PTS-Informationsaustausch Vorgang über die gesamte Betriebszeit der Anlage fortgesetzt wird, wird empfohlen, auch weiterhin den Anlagenbetrieb mit einem fortlaufenden Informationsaustausch zu begleiten.

Weil die vorgetragenen Informationen zu RPVI die kalte Überdruckbeanspruchung und die Stillstands-Situationen nicht explizit einbezogen haben, werden diese Punkte hier auch nicht kommentiert.

Die empfohlen Themenkreise sind folgende:

- Die Behandlung zusätzlicher kritischer Bedingungen wie den Ausfall der Fremdversorgung,
- der zeitliche Rahmen der berechneten Sequenzen – einige Transienten wurden nicht über ausreichend lange Zeiten gerechnet, so dass ein Druckanstieg während des nachfolgenden Störfallverlaufs nicht mehr berücksichtigt wurde -,
- die Berücksichtigung aller relevanten Rissgrößen und Risslagen mit der Bruchmechanik, sowie Stabilitätsbetrachtungen (kleinere Risse könnten wachsen und im nachfolgenden Störfallverlauf instabil werden),
- der Versprödungsfortschritt, sowie die ergriffenen Gegenmaßnahmen, sowie die Verifikation der aktuellen RPVI und von deren Konsequenzen.
- Es wäre von Interesse, die im Rahmen der verschiedenen während des ExpertInnen-Treffens zitierten [BRUMOVSKY 2004a] Programme (qualification test, extended acceptance test, lifetime evaluation programme) ermittelten Werkstoffcharakteristiken mit den Daten aus dem Bestrahlungsprogramm zu vergleichen, um die Streuung der Werkstoffkennwerte auswerten zu können.
- Für die Zukunft ist es von Interesse, Informationen über die Ergebnisse der Bestrahlungsprogramme für beide Blöcke zu erhalten. Besonderes Augenmerk sollte den Bestrahlungsdaten der Proben aus dem Schweißgut der Verbindungsnaht Nr. 4 (inklusive der Wärmeeinflusszone) gelten. Die ersten Ergebnisse von Bestrahlungsproben aus dem Behälter, der im Mai 2004 entnommen wurde, sind 2005 zu erwarten.
- Es wird empfohlen, dass die Ergebnisse der Bestrahlungsproben, die in Block 2 bestrahlt werden (insbesondere die Schweißgutproben der Verbindungsnaht Nr. 4/Block 1 und Block 2, inklusive WEZ) in den zukünftigen Informationsaustausch mit besonderem Gewicht während der nächsten Jahre verfolgt werden. Gleichzeitig wäre es wünschenswert, Informationen darüber zu erhalten, ob die Schweißnaht Nr. 2 in die PTS-Überlegungen einbezogen werden wird.
- Die experimentelle Auswertung der Neutronenversprödung der ETE-Werkstoffe im Rahmen des Bestrahlungsprogramms sollte im Rahmen anhand kontinuierlichen Informationen verfolgt werden, hinsichtlich der Berücksichtigung der im Rahmen der VERLIFE-Methodik vorgeschriebenen Temperatur-Sicherheitsabstände (obere Einhüllende der strahlungsinduzierten T_k-Verschiebung für die Anwendung in der RDB-Lebensdauerbestimmung).
- Die Abminderung der Versprödung des RDB ist von größter Bedeutung für die Druckgefäßintegrität, deswegen sind Veränderungen des Reaktorkerns und auch von Veränderungen der Wiederbeladungsanordnung nach einem der nächsten Reaktorbetriebszeiträume vorgesehen. Die bisher zur Verfügung gestellten Informationen sind bisher zu grob; daher fördert dies weiters Interesse.

Der zukünftige Informationsaustausch sollte auch folgende Punkte behandeln:

- die Stutzenanschlüsse der Hauptkühlmittelleitung,
- die Durchführungen der Kontrollstäbe im RDB-Deckel,
- die Durchführungen für die Kerninstrumentierung und andere technische Leitungen,
- die Dichtigkeit der Hauptflansche, und
- die wichtigsten Beiträge zur Minderung der RDB-Integrität durch Umgebungseinflüsse und andere Schädigungsmechanismen, wie Kühlmittelchemie, Wasserstoffdiffusion, Korrosion, zyklische Belastung, Verhalten bei schweren Unfällen, sowie die Erhaltung und Überwachung der für die LBB-Anwendung erforderlichen Maßnahmen, und die Leckdetektierungsinstrumentierung,
- der Schädigungsfortschritt und die ergriffenen Gegenmaßnahmen bezüglich RPVI-Verifizierung Konsequenzen,

da diese für die RDB-Integrität wichtigen Themen beim ExpertInnen-Treffen nicht behandelt worden sind.

Abschließende Feststellung:

Die tschechischen Experten haben die VERLIFE-Methodik für den Nachweis der RPVI (strukturelle Integrität des Reaktordruckbehälters) über die gesamte Lebensdauer der Reaktordruckbehälter in Temelín verwendet. Im Vergleich zu dem Russischen Regelwerk und den IAEA-Richtlinien wurden durch VERLIFE die Sicherheitsabstände reduziert, und zwar durch VERLIFE-inhärente Vorgehensweisen wie die Reduzierung der postulierten Risstiefe, Reduzierung der Sicherheitsfaktoren, nicht-konservative bruchmechanische Annahmen, usw.

In Kombination mit anderen Unsicherheiten, wie der Modellierung der TH-Transienten, Modellannahmen der Mischungsrechnungen, Werkstoff- und Versprödungseigenschaften, Fluenzmessung, NDE-Zuverlässigkeit, usw., können die resultierenden globalen Sicherheitsabstände nicht als ausreichend angesehen werden. Daher empfiehlt das österreichische ExpertInnen-Team,folgende maßgebende Punkte weiter zu verfolgen, die für die Vervollständigung der VERLIFE Methodik wesentlich sind:

- weitere PTS-Analysen und deren Upgrading
- Auswertung des Bestrahlungsprogramms (beide Blöcke)
- NDE-Verfikationsprogramm für die Integrität
- Fortschritte bei der Verringerung der Versprödung

Die gültige, von SUJB akzeptierte Revision der Spezifikation zum Nachweis der Reaktordruckbehälterintegrität auf Basis der VERLIFE-Methodik wäre ebenfalls von hoher Priorität.

TABLE OF CONTENTS

1 INTRODUCTION

1.1 Background of the project

The Republic of Austria and the Czech Republic have, using the good offices of Commissioner Verheugen, reached an accord on the "Conclusions of the Melk Process and Follow up" on 29 November 2001. In order to enable an effective use of the "Melk Process" achievements in the area of nuclear safety, the ANNEX I of this "Brussels Agreement" contains details on specific actions to be taken as a follow-up to the "trialogue" of the "Melk Process" in the framework of the pertinent Czech-Austrian Bilateral Agreement. To enable an effective "trialogue" follow-up in the framework of the pertinent Czech-Austrian Bilateral Agreement, a seven-item structure given in ANNEX I of the "Brussels Agreement" has been adopted. Individual Roadmap Items are linked to:

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

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

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

The signatories agreed that Czech and Austrian Experts would also regularly monitor the implementation of the said measures jointly within the Czech-Austrian Bilateral Agreement.

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

This „Roadmap" is based on the following principles:

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

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

The objective of the Roadmap process covered by the Roadmap Item 3 as stated in ANNEX I of the "Brussels Agreement" is: *"Reactor Pressure Vessel Integrity and Pressurised Thermal Shock"*.

With the associate objective:

"The reactor pressure vessel (RPV) integrity under pressurised thermal shock (PTS) conditions shall be maintained with a sufficient safety margin against brittle fracture throughout the NPPs service life."

ANNEX I provides the following statements regarding the **"Present Status and specific Actions Planned"**:

"The NPP Temelín is commissioned and operated respecting pressure-thermal (PT) curves calculations developed according to Westinghouse methodology. These calculations will be expanded with set of the further PTS analysis for both units using a step-by-step approach with full respect of the IAEA Guidelines for the PTS analysis. The PTS analysis will be finished in accordance with approved project work plan for this item."

The Roadmap specified that the related Specialists' Workshop would be held in the first half of 2004 to discuss this issue. This workshop on the "Roadmap-Item No. 3" was conducted in Prague on 24 and 25 May 2004 according to Article 7 (4) of the Bilateral Agreement of the Exchange of Information on Nuclear Safety. This workshop was the key element in the monitoring process. In a series of presentations, the outline of the technical approach to the RPVI/PTS Roadmap Item was described by Czech Experts, including the legal framework for the issue and the information provided to the Licensing Authority about the technical approach. The analysis of information made available there played a significant role in the development of the basis for the Preliminary Monitoring Report. The Czech presentations at the Specialists' Workshop covered a broad scope of aspects related to the development and implementation of RPVI/PTS avoidance and mitigation measures. [For the individual presentation titles see under Specialists' Workshop (PM3) on page 52 below].

This Workshop's presentations touched almost all topics and items, which were of interest to the Austrian Experts' Team, except for those, which were treated at a supplementary workshop on the "Roadmap-Item No. 3" this time conducted in Řež on October 7, 2004. [For the individual presentation titles see under Supplementary Workshop on page 53 below].

On behalf of the Austrian Government the Umweltbundesamt (Federal Environment Agency) committed a Specialists' Team composed of international experts to provide technical support for the monitoring of the implementation on the technical level of the RPVI – PTS Issue as listed in ANNEX I of the Conclusions of the Melk Process and Follow-up. This specific technical project is referred to as project PN9 comprising altogether seven predefined "project milestones" (PMs).

The approach to RPVI/PTS avoidance and mitigation at the Temelín Nuclear Power Plant is to rely on a systematic process, which has been established for the development, implementation and maintenance of PTS related options. The presentations provided insight into the extensive work accomplished by the plant operator and its technical support organizations to consolidate the RPVI/PTS issues' resolution. The descriptions identified the approach taken, but as overviews, they provided only limited insight into the results and how these were obtained. Consequently, both sides agreed that the pertinent Czech-Austrian Bilateral Agreement is the appropriate framework giving the opportunity for further discussion and sharing additional information on these issues. From technical point of view, the assessment of the RPVI/PTS issues addresses all the elements and aspects, which are recognised important.

These include supporting accident analysis, assessment of plant vulnerabilities, selection of PTS management strategies, evaluation of plant equipment and instrumentation and the related staff training and qualification.

The project includes all related activities such as the identification of information sources for plant specific data, which are needed for the assessment, analysis of the reference material provided by the plant, and evaluation of the current plant status against the state-of-the-art practice. Gathering appropriate information on the plant status with regard to the above-indicated areas is an essential part of the project. The main concept implemented in the project was to break down the overall subject into the line items, which could then be verified for completeness and compliance with the accepted international practice.

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

The second step (Project Milestone 2), the Specific Information request (SIR) considered to contain the kind of information required for providing profound answers to the VLIs.

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

Project milestone four includes the preparation of a preliminary monitoring report (PMR) on the status of RPVI and PTS at Temelín (PM 4). This task was conducted based on the results of the Prague workshop. This report is also intended to identify RPVI and PTS related issues of further interest. At the time of the Specialists Workshop, the PTS related activities were in the process of implementation in accordance with the planning for 2004.

Therefore, further monitoring is recommended to focus in detail on the continued implementation process, including further attention to be paid to some specific plant design changes implementation. The continuation of the monitoring tasks (project milestones 5 – 7) concentrated on the consolidation of findings and their presentation in the form of the final reports (Final Monitoring Report and Summary Monitoring Report).

1.2 Scope of the project

The project PN9 „Reactor pressure vessel (RPV) integrity and pressurised thermal shock (PTS)“ deals with the topic of RPV damage as a consequence of a thermal shock transient. In the case of most critical transients, the primary circuit is under high pressure. This is one of the main concerns within the reactor safety analysis since the RPV pressure retention and radioactive inventory retention functions are of non-redundant nature by design. A rupture of this component would therefore induce a catastrophic accident.

Consideration of RPV (reactor pressure vessel) integrity as well as the exclusion of the PTS (pressurized thermal shock) at the NPP (nuclear power plant) is an essential ingredient of the defence in depth approach and therefore of utmost importance to Austria.

PTS events should have very low frequencies, since they may have significant consequences resulting in failure of at least one entire barrier (the primary coolant system). As PTS has not been explicitly considered in the design of many older nuclear power plants, which are currently in operation, considerable efforts have been devoted already by most of those plants to prevent PTS events during plant operation, but also at zero-power, shutdown and during outages. PTS prevention has been recognised an important safety issue and is consequently addressed in a comprehensive and systematic way.

In applying current safety philosophy, the consideration PTS and RPV integrity in NPPs usually includes the following elements:

- Identification of event sequences that could lead to PTS;
- Consideration of existing plant capabilities, to avoid PTS events in all operational states of the plant.
- Evaluation of potential changes to the design and/or operation of the plant, which could either reduce the likelihood of PTS events or would mitigate their consequences to the RPV;
- Establishment and analysis of representative and bounding sequences of events that may lead to PTS of the RPV.
- Evaluation of the temperature and stress fields in the RPV wall induced by the PTS event and calculation of the load path on postulated cracks in the vessel region close to the active zone resulting in the identification of critical fracture temperature for the selected event.
- Monitoring of RPV neutron irradiation induced material degradation (embrittlement) and determination of the actual material state in order to evaluate the safety margin with respect to the critical fracture temperature.
- Reduction of the neutron irradiation on the RPV wall by specific core arrangements (i.e. dummy elements, etc.) to limit the neutron induced embrittlement of the RPV steel.

PTS prevention is intended to avoid the escalation of an event into a severe accident. It focuses on the mitigation of consequences of over-stressing of the RPV, by keeping the RPV within the allowable load limits. In general, PTS prevention comprises measures and actions undertaken to ensure that cool-down of the RPV is performed following predetermined procedures, which would assure that the RPV wall temperature gradients are limited to avoid high thermal strains which may cause RPV wall crack overload and failure. Such PTS prevention concepts were adopted in many nuclear plants starting already in the 1980ies.

The Temelín NPP, originally of Soviet design, and later upgraded to include elements of western safety concepts and western equipment, has addressed PTS and RPV integrity late in the construction phase (Russian and Western Codes request pre-service PTSA). During the experts' meetings in the frame of the Melk process, it appeared that the process of PTS prevention implementation at Temelín was late and still not complete, although at the time of the ETE start-up phase covering rules existed ([IAEA 1997], [PNAE-G7-002-86], [SÚJB 1998]) to perform a comprehensive PTSA. The availability of information on the details of the approach adopted at the Temelín NPP was insufficient. Therefore, PTS remained one of the Roadmap Items to be addressed during the follow up to the Melk process. This established the basis and defined the scope of the project.

1.3 RPVI Technical Background

The Temelín nuclear power plant (NPP) is a two-unit facility designed as WWER-1000/320 pressurised water reactors originally according to the standards of the former Soviet Union. Following the political changes, the plant design was upgraded (including redesign of fuel and instrumentation & control equipment delivered by Westinghouse) and put into operation beginning with Unit 1, which had its start-up testing in 2001. In the plant, safety analyses reports (SARs), the plants' response to "design basis accidents" (DBAs) are evaluated assuming a single active failure in the safety system response, and the performance of the plant is evaluated to ensure that basic safety criteria are met. Such SAR assessments are performed more recently also for cases with possible impact on RPVI resulting from PTS events. RPVI issues pertaining to eventual failure under SA conditions however are treated together with SAM considerations and as such the consequences of these accidents – results of international research and development programs – and the issue of accident management (AM) gained prominence in the 1990s. The SA related measures are not discussed within the scope of the PN9 monitoring.

RPVI also involves pre-planning and preparatory measures for guidance and procedures, equipment modifications to facilities procedure implementation, and PTS management training.

The overall objective is to further reduce the risks RPV failure. It is the responsibility of the licensees to develop and implement a PTS management program. RPVI/PTS management plays an important role in the defence-in-depth concept. Design verification as well as components and system functions assurance have been performed, also for all items, which have been added to the design of the plant in order to enable it to cope with, to prevent or to mitigate PTS events and their adverse effects on RPVI and the consequences, which could be severe accidents. When the Czech Republic and the Republic of Austria jointly issued the Melk Concluding Statement and the Road Map, the issues of RPVI and PTS and their management were specified for further technical exchange.

1.4 General description of the project concept

To appreciate this development, a team under the Technical Project Management of IRR-ARCS performed the monitoring work on this project. The team addressed in the first place a broad 'horizontal' view of the general assessment of Temelín RPVI and PTS based on underlying analyses and principles.

<u>Horizontal segment</u>

The assessment of principles, standards and practices is aimed to discuss the Czech regulations and guidelines used for RPVI demonstration in the context of the Russian Code requirements under which the RPV has been design and constructed, and the IAEA Guidelines that were elaborated especially for WWER reactors RPV lifetime assessment. The comparison with Western state-of-the-art is included for the specific issues of RPVI in order to provide an insight into Western safety philosophy and practice and to judge whether the Czech approach and measures taken can be considered comparable to Western practice.

<u>Vertical segment</u>

Secondly a team of TSOs looked after possible impairments of the WWER 1000 RPVI based on generic assumptions in a set of peer review like spot-check analyses of the vulnerability of the RPVs in a 'vertical evaluation'. The technical work was managed in parallel by providing transfer of information and joint discussion of important issues. The FMR is the responsibility of IRR-ARCS and gives credit to the findings from both, the 'horizontal' and the 'vertical' evaluations.

The potential for loss of RPV integrity due to overcooling and subsequent re-pressurisation (pressurised thermal shock) for Temelín NPP was selected for monitoring. RPV material ageing caused by neutron embrittlement was to be considered. Two scenarios with elevated potential for RPV failure with possible containment failure consequences, which could result in specific safety concerns to Austria, were selected for bounding case monitoring. This segment does also include the collection of information on the Temelín RPV embrittlement behaviour over time, as well as the vessel's material history and usage and its thermal shock vulnerability.

Modern analytical techniques have resulted in integrated computer code applications. More advanced simulation options in this field use three-dimensional models and integrated codes that are able to combine fluid dynamics and heat transfer applications, and employ stress analyses codes and fracture mechanics tools. The monitoring process was concentrated on the engineering approach taken by ČEZ to have the RPVs licensed by the SÚJB (State Office for Nuclear Safety).

The assessment of principles, standards and practices is aimed to discuss the Czech regulations and guidelines used for RPVI demonstration in the context of the Russian Code requirements under which the RPV has been designed and constructed, and the IAEA Guidelines that were elaborated especially for WWER reactor’s RPV lifetime assessment. The comparison with Western state-of-the-art will be included for the specific issues of RPVI in order to provide an insight into Western safety philosophy and practice. Most Western countries are bound to the U.S. ASME Code or at least have adopted main parts into their National Codes (Germany, France). In contrast to this proceeding, the United Kingdom has a non-prescriptive Code that is based on safety principles. Some of these safety principles are included into the description of Western state-of-the-art to demonstrate the different practice.

1.5 Assessment Framework

The assessment of the effectiveness of the prevention and mitigation of PTS at Temelín has been performed in the context of several activities. In the main, these activities have involved an assessment of the state-of-the-art in RPVI/PTS in Western Europe (and more broadly including the US), an assessment of the performance of a generic WWER 1000 NPP in PTS events, taking into consideration Temelín-specific plant characteristics (to the extent possible derived from available information) by means of state-of-the-art combined code calculations.

One two-day Specialist Workshop was held in Prague at which presentations on various aspects of RPVI/PTS management were made by Czech Experts and discussed with the experts of the Austrian delegation, including members of the PN9 Technical Project Management and Team.

A supplementary one-day Specialist Workshop was held in Řež, at which Czech Experts made presentations on various aspects of RPVI/PTS management with regard to material properties, training and ISI. These presentations were then discussed with the experts of the Austrian delegation, including members of the PN9 Technical Project Management and Team.

It should be clear from the beginning that the assessment is not based on a review of original Temelín documentation, such as the PTSA, the related Emergency Operating Procedures (EOPs), the material’s and surveillance specimen’s data bases, including component production, Pre-service and In-Service-Inspections procedures and results, as well as the VERLIFE provisions ad the required training of staff – all of this was not provided for review.

Similarly, although Czech Experts performed a number of PTS calculations as technical support for the RPVI demonstration, the details of these calculations were not available for review. Likewise, the updated Pre-Operational Safety Analysis Report (POSAR) was not available for review.

Finally, the plant itself was not available for detailed confirmation of geometric arrangement and other relevant details.

The Team has had in the past the opportunity to review the POSAR and the PSA documentation, and has had the opportunity to discuss with Czech Experts over the past three years in which the Melk Protocol and Road Map activities have taken place. Thus, the PN9 approach is to capitalize on this experience, on PTS calculations made with state-of-the-art codes, on knowledge of the state-of-the-art in RPVI/PTS management in Western Europe and the United States, and on knowledge gained over the years about the Temelín plant design and systems technology as well as RPVI/PTS management programs under way. One basic fact is that rules and regulations applied to Temelín NPP for the design, construction and design verification as well as for the implementation of RPVI/PTS were combined from rules and regulations originating from Russia, Czech Republic, the United States and IAEA documents. Large part of these combinations were harmonized and issued as recommendations in the frame of the VERLIFE project, and the regulatory body has accepted the outcome.

The IRR-ARCS team addressed possible impairments of the WWER 1000 RPV integrity function of the Temelín NPP due to adverse cooldown procedures under accident conditions. Investigations in that area, which focuses on in-depth analysis of the specific vulnerability of the RPV, are still going on.

Some topics, which relate to RPVI verification at the Temelín plant, have been excluded from treatment here, since the specific information is provided in the PN10 report [see List of Austrian Projects, ANNEX C].

Identification and evaluation of results published on WWER-1000 material properties with respect to conservative predictions of the neutron embrittlement

The steel used for the WWER-1000 RPVs was supposed to be less radiation sensitive than the WWER-440 mild RPV steel. The embrittlement of the material due to neutron irradiation has been studied in the former Soviet Union using irradiation of samples in test reactors (high neutron flux allowing investigations of neutron fluences covering the full service lifetime of the RPV). The results were implemented into the Russian Code formula as specified embrittlement factors; this formula allows the prediction of neutron embrittlement during NPP operation. These neutron embrittlement data from specimen irradiated in test reactors showed already that the aim of developing RPV steels with low 'radiation sensitivity' was not reached.

After commissioning of WWER-1000 reactors, it turned out that the specified embrittlement factors might not be conservative. It was also realised that the WWER-1000 surveillance programmes had severe deficiencies that did not allow reliable information on the neutron embrittlement progression of the plant-specific material.

Part of the project was therefore to analyse published embrittlement data of WWER-1000 materials with respect to the conservatism of the embrittlement factor specified in the Russian Code.

PTS analyses for selected bounding cases

In the frame of preparation of the discussion of PTS analyses for NPP Temelín expected to be presented during the Workshop in 2004, the Austrian Experts' Team assisted by external technical support institutions (TSOs) performed PTS analyses on selected bounding cases.

Pressurized Thermal Shock (PTS) is a concern under two distinct sets of circumstances, which can occur as a result of a loss-of-coolant-accident (LOCA):

- Asymmetric cooling of the downcomer RPV wall metal at high pressure and
- Re-pressurisation of the system after cooling below the transition temperature between brittle and ductile behaviour in the downcomer RPV wall metal.

For small breaks, the first type of PTS issue provides the dominant cause for concern. In such cases low loop flows resulting from a loss of off-site power are possible and allow thermal stratification in the cold legs. This stratification then can lead to a cold plume of HPI (high pressure injection) water cascading down the downcomer wall at high pressure.

For intermediate to large breaks, the first issue is not important since the system depressurises quickly due to the large mass inventory loss through the break. The second issue, however, is a concern since larger breaks, which assume no loss of offsite power and are isolatable, will cause a significant cooling of system due to the full capacity HPI flow. This full flow will also cause a significant and rapid re-pressurisation of the RCS (reactor cooling system) as soon as the system is refilled following break isolation.

The inadvertent opening and remaining stuck open (SO) of the PRZ PORV (pressuriser power-operated relief valve) is a transient, which combines both effects: it is a large compensated break and if LBB (leak-before-break) will be applied to the loop pipe, this is the largest break, which is isolatable. This transient was chosen as one of the bounding cases. Several variants of this transient were defined by varying the boundary conditions (the number in the identification indicates time of PORV re-closure):

- Case V00: PORV-SO-1800
- Case V01: PORV-SO-1800
- Case V02: PORV-SO-0750
- Case V04: PORV-SO-0750
- Case V05: PORV-SO-0750
- Case V06. PORV-SO-0750

HZP (hot zero power) cases: V00, V01, V02, V04, V05

V00, V02: two of the three TQ3[19] pumps are injecting water in the primary circuit

V01, V04, V05, V06: all three TQ3 pumps are injecting water in the primary circuit

The other bounding case transient selected to represent a large break LOCA was the Double-Ended Guillotine Break of primary coolant pipes (ND 850) near the reactor vessel inlet or outlet nozzles. Two cases were analyzed:

- Case V07: Cold-Leg Large Break LOCA
- Case V08: Hot-Leg Large Break LOCA

For the core cooling function the timing of the blowdown end, the beginning of reflood and quench, and the timing and magnitude of the peaks during cladding temperature history in the blowdown and reflood phases were of interest.

- The break size in the primary circuit is very important, because it affects the coolant loss break flow and may lead to fuel rod heat up; the hot leg rupture is less conservative than cold leg rupture, if core-cooling aspects are considered.

The thermal-hydraulic transient analyses have been performed with the RELAP5/3.2 code.

The phenomena associated with high-pressure safety injection (HPI), and associated stratification/cooldown effects, in cases where this injection is under loop flow stagnation conditions, have received considerable attention since about 1981. The Regional Mixing Model (RMM) has been successfully employed for modelling and the associated computer pro-

[19] High-Pressure Injection System

grams REMIX and NEWMIX for the simulation part of WWER-1000 RPV. Both codes are suitable for predictions for the Integrated Pressurized Thermal Shock (PTS) study.

More recently, a number of utilities in the Czech Republic, in Finland and Japan were requested to perform PTSAs and made use of REMIX/NEWMIX for similar purposes. Thus, the Experts' Team used also this code for their bounding case calculations. For the Czech PTSA the REMIX/NEWMIX code was applied in the case of "cold plumes" and the CATHARE code in the cases of "cold sectors" and "cold stripes".

Structural/fracture mechanics analyses were performed using the 3D FEM Code ANSYS and analytical approaches in accordance with the Russian Methodology.

1.6 Specialists' Workshop (PM3)

Several tasks had to be performed by the Experts' Team for the preparation of the Specialists' Workshop:

- Identification of the requirements for RPVI in accordance with the state of the art practice.
- Identification of information available on RPVI activities at the Temelín NPP.
- Identification and evaluation of results published on WWER-1000 material properties and behaviour with respect to conservative assessment.
- Provision of the Verifiable Line Items.
- Provision of the Specific Information Request.
- Preparation of the Briefing Material, Briefing Session
- Preparation of PTSA for selected bounding cases.

The results of all tasks performed have served for monitoring the actual state of the Temelín NPP, the preparation of the Workshop and the introduction into the various disciplines for the Austrian delegation.

The Specialists' Workshop scheduled in the frame of the "Conclusions of the Melk Process and follow-up" for the first half of 2004 took place at SÚJB in Prague during May 24th and 25th, 2004.

The Agenda of this Workshop covered the following presentations:

P. Krš	Welcome and Introduction of Meeting Participants (Czech Delegation)
G. Polte	Welcome and Introduction of Meeting Participants (Austrian Delegation)
J. Žďárek	Reactor Pressure Vessel Integrity (RPVI) Assurance Approach
M. Šváb	SÚJB Comment on Current Legislation Basis on the RPVI/PTS
M. Brumovský	VERLIFE Methodology with respect to the RPVI and PTS
J. Žďárek	QA Programme for Analysis, Assessment and Related Support Activities
M. Sýkora	EOP Strategy for PTS and Relevant Systems
V. Mečíř	RPV Fluence Minimization
V. Pištora	Comparison of IAEA, Russian and VERLIFE Methodologies
P. Král	Selection of Scenarios for PTS Analyses and TH Methodology
P. Král, P. Mühlbauer, M. Malačka	Overview of TH Analyses Results for PTS
J. Shejbal	Statement on the NDE Qualification Process

L. Horáček	Qualification of NDE Respective to the “PTS” Affected Area of the RPV, including results for the beltline welds
A. Kačor	Integrity Models Description
V. Pištora	Summary of Results from PTS Integrity Evaluation
J. Žďárek	Surveillance Programme Status
M. Holan	UJE Position
M. Šváb	SÚJB Position

The monitoring evaluation of the Czech contributions is integrated into the following chapters of the Preliminary Monitoring Report.

Supplementary Workshop

The Supplementary Specialists’ Workshop held in Řež, October 7, 2004 on the occasion of the PN10 Roadmap Workshop there (October 7 to 8, 2004) answered remaining questions with the Agenda of this Workshop covering the following topics with PN9 related presentations:

P. Krš	Welcome and Introduction of Meeting Participants (Czech Delegation)
G. Polte	Welcome and Introduction of Meeting Participants (Austrian Delegation)
M. Brumovský	Material Problems in PTS
V. Pištora, P. Král	Update of PTS Results – TH Analysis
V. Pištora, P. Král	Update of PTS Results – Structural Analyses
V. Pištora	Sensitivity Analysis on the Influence of Extent of RPV FE Model (180 ° or 360 °)
M. Sýkora	PTS Training Programme Elements
M. Šváb	SÚJB Position (partly also applicable here)

In a final statement related to these presentations concerning the PN9 topics the Austrian delegation welcomed the additional information provided, as well as the opportunity to obtain answers to further questions.

1.7 Bilateral Meeting

The 13th Bilateral Meeting under the Agreement between the Government of Austria and the Government of the Czech Republic on Issues of Common Interest in the Field of Nuclear Safety and Radiation Protection took place in Dolni Dunajovice, on 29-30 November 2004. On this occasion, the preliminary results of the monitoring were presented to the Czech delegation and the replies were discussed [ANNEX F]. Since there were no additional Items to be treated in association with the Monitoring of Roadmap Item 3, there were no results obtained for this topic.

An overview of the activities that will follow the Bilateral Meeting was given and further information on the issues associated with RPVI and PTS is envisaged be treated in the upcoming Bilateral Meetings.

1.8 Structure of this report

The evaluation of all additional information provided by the Czech Experts during the Bilateral Meeting and additional results of pilot studies conducted were incorporated into the Preliminary Monitoring Report to transform it into this Final Monitoring Report.

Sections 2 to 8 *(This report will only consider the actual status of the foregoing projects PN2, PN3, PN7, where required)* provide a comprehensive evaluation of relevant aspects relating to RPVI/PTS management the PTS related program at Temelín. The material presented in these sections is arranged into several subsections corresponding to the selected evaluation factors or aspects. With some exceptions, each of these subsections comprises of three parts:

- *"Description of the issue and fundamentals",*
- *"The current state-of-the-art requirements and practices",*
- *"Current plant status" and*
- *"Evaluation".*

The first part provides an informative introduction to the specific issue; the next part defines the 'assessment criteria' specific to the evaluation area/factor i.e. the basis to be used for the assessment.

Typically, the *"Current plant status"* part includes a brief discussion of the related plant status with references to other sources of information.

The *"Evaluation"* part summarises the results of the assessment against the "*specific assessment criteri"*.

Deficiencies or safety concerns as well as the recommended issues for further monitoring are identified in this report at the end of each chapter. All these conclusions and issues of further monitoring are summarised in chapter 9.

Annex A summarizes the PTSA concept as conducted for RPVI. (Page #201 ss.)

Annex B provides detailed information on the frequently cited VERLIFE project and intended results in order to give an orientation about the accomplishments reached. (Page #205 ss.)

Annex C lists the Austrian Project identification in combination with the Roadmap Items identified in the "Road Map". (Page #217 ss.)

Annex D lists the Benchmark calculations' types that were conducted by the Austrian Experts' Team. (Page #221 ss.)

Annex E Is a copy of the Specific Information Request as compiled by the Austrian Experts' Team for project PN9, considered to contain the kind of information required for providing profound answers to the VLIs. (Page #225 ss.)

Annex F Provides a copy of the Slides presented at the Bilateral Meeting in 2004 presenting an overview of the Preliminary Monitoring Report's results. (Page #243 ss.)

Annex G Contains the information provided by the Czech Side in order to add to the heat transfer issue discussed with the PTSA. (Page #253 ss.)

Annex H MISSION STATEMENT as adopted by the Austrian EXPERTs' Team.

The present report contains all evaluated information on the topics in question and represents an evaluation of the accumulated knowledge about RPVI/PTS during the Melk Process.

2 REACTOR PRESSURE VESSEL INTEGRITY (RPVI) CONCEPT

Areas of Monitoring

No	VLI/VLI group description
1	**RULES & REGULATIONS**
1.1	**REACTOR PRESSURE VESSEL INTEGRITY AND PTSA RULES, CODES, STANDARDS AND GUIDELINES**
1.1.1	**RPVI and PTSA National requirements**
1	Are there any national requirements on the overall RPVI (Reactor Pressure Vessel Integrity Program) and PTS issues (Pressurized Thermal Shock Program)?
2	Which national requirements have been established and when?
3	Have the national requirements been verified for completeness, comprehensiveness, consistency, and coverage of the needs imposed by the various codes applied and with what result?
1.1.2	**RPVI and PTSA designer and manufacturer requirements**
1	What is the status of fulfilment of the Russian Code requirements concerning design and manufacture at ETE?
2	How do national design and manufacturing requirements correspond to the requirements imposed by the Russian Code?
3	Have requirements other than those imposed by the Russian Code(s) been introduced in the national design and manufacturing requirements? If yes, how do they respect and affect the Russian design features?
1.1.3	**RPVI and PTSA recommendations in the IAEA Guidelines**
1	In what respects are the Temelín RPVI and PTS programs different from the recommendations contained in IAEA-EBP-WWER-08, “Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants”, April 1997? Has a comparison between the Temelín RPVI and PTSA programs with IAEA-EBP-WWER-08 been conducted, and if so with what results? *[Note that Mr. Brumovský from NRI Řež and Mr. Tendera from SÚJB are listed as “Contributors to Drafting and Review” for IAEA-EBP-WWER-08.]*
2	What PTS-relevant operator interventions (e.g., trip of main coolant pumps, throttling or termination of ECCS pump operation, break isolation, initiation of secondary side cooldown, initiation of primary side feed and bleed) have been considered in the Temelín PTSA? *[This issue is discussed in Sections 4.1.6, 4.2.2, of IAEA-EBP-WWER-08.]*
3	How has the timing of operator interventions (of either positive or negative character) been considered in the PTSA?
4	How have uncertainties in the results of the PTSA been addressed for Temelín?
5	Given the unique core design of Temelín – by what means has the uncertainty in fluence been assessed for Temelín and how have the uncertainty bounds on fluence been established and reflected in the PTSA? *(Compared to other WWER-1000/320 cores, the Westinghouse fuel, control rods, chemical composition, etc. are most essential changes in the core design)*

Further topics examined for monitoring were:

No	VLI/VLI group description
1	**RULES & REGULATIONS**
1.2	**COMPARISON OF NATIONAL, RUSSIAN, EU AND US CODES AND PRACTICE**
1.2.1	**RPVI and PTSA comparison with Russian, and Western European State-of-the-Art**
1	Which Russian and European PTSA requirements and rules practice have been followed (explicitly, or by following equivalent national regulations)?
2	Which Russian and European PTSA requirements concerning operational pressure-temperature limits are followed at ETE (explicitly, or by following equivalent national regulations)?
3	What requirements and rules comparable with Russian and European practice are followed concerning surveillance programs?
4	What requirements and rules are followed concerning RPVI/NDT programs comparable with Russian and Western practice?
5	Do the selected strategies for avoidance and/or mitigation of PTS and its consequences reflect the current international knowledge and practices?
1.2.2	**RPVI and PTSA comparison with USNRC acceptability requirements**
1	What, if any, are requirements for PTSA and RPVI at ETE according or corresponding to U.S. Codes and/or was current NRC practice applied (explicitly, or through equivalent national regulations)?

2.1 The RPVI and PTS general concept

2.1.1 Description of the issue – fundamentals

Reactor Pressure Vessel integrity is required in order to prevent excessive loss of coolant from the primary coolant circuit. Even though, small losses are inevitable – mostly through penetrations of the primary coolant pressure boundary, such as the main coolant pump's motor shaft sealing, valve stems, control rod assembly penetration, and foremost the inevitable steam generator tube leaks – the Reactor pressure vessel, its closure and the lid sealing are leak-tight, resisting the internal PCS nominal pressure of around 15,4 [MPa], with a safety margin of up to 50%.

The leakages of the PCS during operation are limited to quantities of around 30 [l/h], and should not exceed these values, in order to allow for reliable leak detection according to the leak-before-break (LBB) concept in case a larger and potentially hazardous leak occurs. (Water at high temperatures and high pressure transforms immediately to steam when released to the environment, in this case the containment free volume. Instruments available can detect such steam quantities).

Leakages are limited and so are leaks – deficiencies in some way allowing the pressure boundary to be bypassed. Such deficiencies, flaws and other defects can be of different origin – they may be the result of low quality manufacturing, material and/or component degradation, caused by excessive loads, wear, load cycles, corrosion, erosion, design deficiencies, operation and maintenance errors. Incorrect In-Service-Inspection (ISI), corrective actions and quality assurance are also found as root causes for deterioration of the primary circuit integrity.

Excessive leakage can indicate a small-break-loss-of-coolant accident (SBLOCA) initiation with the implication that the principal safety function to "cool the fuel" could be at stake.

In case the leak location is at the RPV outer wall, the situation could turn out to be particularly critical, since any leak below the top of the core has as an implication the possibility that the core could be uncovered in the long run.

Furthermore, large leaks at the RPV wall tend to disturb the flow and the flow regimes in the entire vessel in such a way that cooling of the fuel might be impaired in some areas.

The design base contains also a defined value for the maximum allowable nominal leak diameter, meaning the circular “replacement leak’s” diameter causing the same coolant loss flow out of the RPV, as the actual leak. These leak dimensions are typically in the range of several tens of square centimetres, like 50 ÷ 80 [cm^2] in some cases.

Therefore, the dimensions indicate that already very limited leaks are most likely to exceed safety margins in such an accident sequence.

The situation had to be regarded as critical once it became known that assumptions about the ductility of the RPV materials over service time could not be proved true. Due to excessive neutron embrittlement during operation of some of the plants especially in the US and Russia the material properties at the RPV weldments degraded to such an extent that brittle fracture in combination with high speed crack propagation had to be considered possible.

Extended longitudinal as well as circumferential cracks could be the consequence of excessive loads from operation or resulting from accidents. The search for loads likely to exceed design limits revealed candidate load collectives in particular in the area of non-self-equilibrating thermal strains, resulting in stress fields largely exceeding the material properties’ provisions made in the design.

The thick wall of the RPV makes it susceptible to brittle fracture if ever rapid cool-down of this wall occurs when large quantities of coolant with low temperature are fed into the RPV via one or more of the inlet nozzles from the cold leg of the PCS (the pressure side of the MCP). The most prominent candidates to cause a loss of RPV integrity are therefore in this context:

- Cold over-pressurization events
- Pressurized thermal shock (PTS) events

Other types of loss of RPVI are related to material degradation due to stresses in combination with environmental effects like corrosion, cycling fatigue etc. However, the focus of the work to be accomplished within this project is on the PTS consequences, even though assumptions about deficiencies and the related defects’ fracture mechanics behaviour are basically of the same nature.

In any nuclear fission power plant, nuclear fission takes place in the reactor core. The core contains the nuclear fuel, and is confined in a thick-walled steel reactor pressure vessel. During the process of nuclear fission, neutrons are generated and heat is transferred to the water coolant. In a pressurized water reactor (in which boiling of the coolant is prevented by maintaining the system at a pressure above evaporation limit), an overpressure of 15,4 [MPa] and a coolant temperature of about 300 [°C] are sustained during operation. The heat in the coolant is transferred while passing the coolant through the steam generators, which conveys the steam produced in the secondary side of the steam generators to a steam turbine, which powers an electrical generator.

As indicated above, the reactor pressure vessel (RPV) is exposed to high pressures and temperatures during operation. Near the active zone of the reactor core, the neutrons produced by nuclear fission bombard the reactor pressure vessel steel. This includes the weld in the area of the reactor core as well as the entire wall there.

The pressure and temperature loads and also the neutron flux require a consistently high quality of base material of the RPV and the weld materials. Sufficient safety margins must be guaranteed during the entire lifetime of the reactor since the potential risk of rupture of the

RPV can eventually not be compensated by the containment retention capability. This could result in serious radiological consequences for the environment.

The bombardment of the wall of the RPV and the weld in the core area by high-energy neutrons during operation leads to a progressively higher susceptibility to embrittlement of the vessel metal and weld metal. This might result in brittle fracture of the RPV (i.e. RPV rupture) due to unexpected operation conditions with cold-water-injection and accident pressure build up. A plume of cold water developing very fast from top down along the reactor pressure vessel wall endangers RPVI. The material – ductile at high temperatures – behaves brittle below a specific transition temperature.

Rapid cooling of the RPV and the weld below this transition temperature, followed by pressurization, increases the risk of brittle fracture. The transition temperature from ductile to brittle behaviour (ductile-brittle transition temperature DBTT) is characteristic for the individual material and changes due to ageing (neutron irradiation damage, thermal and mechanical fatigue). Increasing neutron irradiation damage induces an increase of DBTT. It is necessary to ensure that the transition temperature T_k of the material never, even not when approaching the 40 years lifetime, reaches a critical limit[20] T_k^a -, which results from the thermal-shock analyses. During the operating lifetime of the reactor, it is necessary to guarantee that brittle fracture cannot occur. This guarantee must comprise not only normal operations but also any kind of incident.

2.1.2 State-of-the-art requirements and regulations

It is the state-of-the-art that before start-up of a reactor, a prognostic estimate of the process of embrittlement based on the related standards must be accomplished. All possible operational states and incidents must be investigated with a view towards guaranteeing brittle fracture safety.

In the larger context the emerging issues and further development in RPVI and PTS on the European level is focussed to catalyse the effort of European key players in NPP plant life management in the SAFELIFE Action. The support will be based on the successful and well-established European Networks AMES, NESC (network for the evaluation of structural components), ENIQ (European network for inspection qualification) and on more recent ones such as NET (neutron evaluation techniques), AMALIA (Assessment of Materials Ageing under the effect of Load and IASCC) and SENUF (Safety of Nuclear Installations).

RPVI requirements

The continued assurance about RPVI throughout lifetime of the plant is based on a concept that includes the following issues:

- PTSA (pressurised thermal shock analysis): the calculation of loading paths for selected critical accident transients, determination of the critical value of DBTT (i.e. of the reference temperature as defined in the National Codes).
- Surveillance program: determination of the neutron embrittlement using RPV specific samples[21] in irradiation capsules inside the RPV with a lead factor[22] of about 2.
- Non-destructive testing programmes to assure the continuous integrity of the RPV with respect to flaws, cracks and other defects.

[20] For WWER RPVs this temperature is called “maximum allowable critical temperature of brittleness T_k^a“.

[21] The samples are made of the same steel charges and weld materials as the RPV, are manufactured under identical conditions and have experienced the same heat treatments.

[22] Relation between the neutron flux at the irradiation capsule position and the neutron flux at the wall in the belt-region.

- Mitigation measures such as neutron minimization (design of the core, implementation of dummy elements, etc.)
- Emergency operational procedures (EOPs) for PTS events.

RPVI as covered by the Czech side

Workshop presentations:

M. Brumovský, J. Žďárek: Reactor Pressure Vessel Integrity (RPVI) Assurance Approach;

M. Holan: RPVI and PTS – UJE Position.

The components for WWER-1000/320 NPPs were designed based on [OPB 1973], and the associated standards/rules available at the design stage, and in parallel with the development of these rules [OPB 1982].

Reactor pressure vessels for the Temelín NPP were manufactured in ŠKODA Nuclear Machinery, Plzen according to the original drawings and technological documentation provided (based on the purchased licence) by the Russian company 0KB Gidropress, Podolsk, General Designer of WWER type reactors in accordance with the Russian Code (see also: [Rez_IAEA 1996]).

At the time of the POSAR compilation and start-up of the ETE units 1 and 2, the following normative documents concerning the PTSA were valid:

- The Russian Standards for Calculations [PNAE-G7-002-86] with a detailed description of the PTSA methodology.
- The IAEA Guidelines [IAEA 1997] with a harmonization of Western and Russian standards for PTS analyses, developed with the active participation of the Main designer and the Czech Experts.
- The national Czech Guidelines [SÚJB 1998].

These are the documents cited in the Melk Agreement as the normative bases for the PTS analyses to be performed for the Temelín NPP and they are defined fundamentals in the POSAR.

These documents include comprehensive requirements, rules and regulations for a preservice PTSA, which was not performed; the limiting pressure-temperature curves calculated according to the Westinghouse concept do not fulfil the cited Standards and therefore they cannot replace a PTSA.

In the frame of the Workshop in 2002 concerning PN2, some results on the PTSA for the Temelín NPP were presented [PISTORA 2002] with comparable, but slightly deviating basic standards:

- The Czech Guidelines [SÚJB 1998] were cited in the first place, together with an announcement of an update.
- In the second place a new methodology (VERLIFE) was introduced, that was to be developed as the result of a EU-project under the leadership of Czech Experts.
- Furthermore, the named documents included the IAEA Guidelines [IAEA 1997] and the Russian Standards [PNAE-G7-002-86].

This fact shows that after the start-up of the plant and having the first PTSA results a deviation form the definite normative basis started; the following will show the reasons and the aim of these deviations:

In 2004 the normative basis defined earlier was abandoned completely, the introductive presentation [BRUMOSVKY 2004a] relies only on the VERLIFE methodology as the basis for the PTSA performed.

The presentation of the Czech Regulatory Body [SVAB 2004] does cite the former normative basis, but at the same time it declares: "The VERLIFE Procedure has been accepted for lifetime and integrity evaluation of components and piping in WWER type NPPs by the SÚJB."

The representative of the Nuclear Research Institute, Řež [BRUMOVSKY 2004a] summarized the RPVI concept as being based on the following steps:

- **Qualification test programme** (standard mechanical tests[23], fracture toughness tests, radiation resistance tests under operating temperatures up to fluences between $9x10^{22}$ and $1x10^{24}$ [1/m^2], thermal ageing tests at temperatures up to 450 [°C] for up to 10 000 [h].
- **Extended acceptance tests** during RPV manufacture on RPV material of „all three plain rings in the core area“ (standard testing, static fracture toughness tests, radiation resistance test under three neutron fluences).
- **Programme of RPV lifetime evaluation:** implemented during 1994 and 1998 by NRI, Řež and ŠKODA Plzen: radiation damage of RPV weld material at various fluences, additional fracture toughness curve for weld material, corrosion-mechanical properties of base metal, weld metal and austenitic cladding in conditions of the primary coolant regime.
- **Modified surveillance programme:** surveillance capsules allow temperatures of the surveillance samples equivalent to the RPV wall temperature, irradiation with a lead factor below 2; fluence monitoring on the RPV wall outside surface; RPV austenitic cladding samples beyond the original surveillance requirements (base metal, weld metal, heat affected zone).
- **NDE programme during manufacturing and operation:** inspections from outside and inside of the RPV (ultrasonic, eddy current, visual inspection); separate quality assurance programmes for all types of RPV inspections
- **PTS calculations:** the calculations are performed in accordance with the VERLIFE methodology.
- **EOP strategy for PTS** as part of the operation and maintenance activities.

According to the representative of the Utility ETE [HOLAN 2004] the Czech RPVI concept is based on:

- Operational provisions for safe operation:
  - RPV fluence optimization – core design
  - PTS events impact mitigation – EOPs
- PTS analytical assessment
- Surveillance program and in-service inspection

With respect to the PTS analysis it was stated in the presentation:

"Structural calculations are performed according to "Methodology of the Structural Part of the PTS Assessment for Temelín NPP and Dukovany NPP" that is based on "Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs, VERLIFE" prepared within the Project of the 5th Framework Programme of the EU."

The VERLIFE methodology was accepted by SÚJB in May 2004. The full documentation on the VERLIFE methodology application is not available for the Austrian Experts. The Austrian Experts obtained limited information on the VERLIFE during the Workshop presentation in 2004 in Prague and the publications [i.e. BRUMOVSKY 2003a, BRUMOVSKY 2003b, BRUMOVSKY 2003c], respectively.

[23] Tensile strength, Charpy-V-notch, hardness, bending

2.1.3 Current plant status

The conclusions of the NPP Temelín representative were:

“Integrated RPV Integrity assurance approach is in place at Temelín NPP which is comparable to the Western state-of-the-art.

Current PTS results does not indicate any transients for which sufficient margin between resulting maximum allowable critical temperature of brittleness T_k^a and highest predicted end-of-life value of T_k for Temelín RPVs would be not maintained.“

2.1.4 Evaluation

The individual steps will be discussed within the following relevant detailed chapters:

- PTS analysis
- Surveillance program – material embrittlement
- Non-destructive testing
- Mitigation measures

A PTS analysis has to be performed according to Code regulations before the start-up as part of the licensing to demonstrate the structural integrity of the RPV throughout the service life.

The NPP Temelín was started without performing a pre-service PTS analysis; the Regulatory Body accepted the operational limiting p-T curves (performed according to the methodology of the Westinghouse concept) as preliminary demonstration of PTS related RPVI.

The Austrian Experts did not consider the operational pressure-temperature limits (Westinghouse concept) as appropriate substitute for a PTS analysis. Besides, the performed analysis was based on non-conservative assumptions.

The Workshop presentation on first results of PTS analyses within the frame of the project PN2 (Conclusions of the Melk Process and Follow-up: Roadmap Item No. 1: High Energy Pipe Lines at the 28,8 m Level) provided first information on the concept of PTSA being performed for NPP Temelín.

The use of optimised steels – not radiation embrittlement susceptible – one prerequisite of state-of-the-art RPV integrity, is not met for Temelín RPVs.

There are doubts that a PTSA performed according to the normative basis as defined at the time of start-up would have supported awarding the operation license.

During the time period of the Melk process and the related discussions on the PTSA issue the normative basis was completely restructured (as can be seen in the following and under the specific topics).

The late completion then of the PTSA however, allowed taking credit from the adoption of significantly less conservative Standards as the acceptance criteria[24].

[24] A similar development can be observed in the Russian Federation: Since all WWER-1000 reactor pressure vessels would not be licensable on the basis of the Russian Norm [PNAE-G7-002-86] or the IAEA Guidelines [IAEA 1997] a new normative approach was developed under the authoritative participation of the Main Designer that reduced the safety margins and safety factors substantially (for instance the postulated crack depth by a factor of 8) in order to allow the demonstration of RPVI under PTS conditions an thus allowing licensing of WWER-1000 reactor pressure vessels.

2.2 Conclusions on Rules and Regulations

The global approach as indicated in the presentations at the Workshop provided for the following monitoring findings:

- The Austrian Experts appreciate that the Czech side is no more considering the operational pressure-temperature limit curves as appropriate demonstration of avoidance of unacceptable PTS sequences.
- The RPVI concept, as it pertains to the PTS analysis approach, appears to follow the state-of-the-art practice and the IAEA Guidelines with respect to analytical methodology. The IAEA Guidelines safety precautions were significantly reduced the way they are interpreted in the new application of the VERLIFE methodology.

Further substantiation is performed in the related chapters of this document. The standards, rules, regulations and in particular recommendations are discussed there.

Although the IAEA Guidelines on PTSA (1997) are part of the Czech legislation and are cited in the "Conclusions of the Melk Process and follow up" as basis for the PTS analyses to be performed, the VERLIFE methodology application has adopted no safety factors in the SIF calculations[25] (whereas the IAEA Guidelines do call for safety factors application[26]), reduced the postulated crack size (not only in comparison with [IAEA 1997] but also with respect to [SÚJB 1998] and allowed the application of the WPS effect with 90% of global maximum of the peak stress intensity factor (as compared to 80% of the peak level as defined in the IAEA Guidelines; in [SÚJB 1998] taking credit of WPS was not included). This is a considerable reduction of conservatism in comparison with the recommendations of the IAEA Guidelines and the Czech Standards [SÚJB 1998] for PTSA.

The document on the VERLIFE methodology application – the basis for the global RPVI concept for NPP Temelín – has not been made available to the Austrian Experts' Team. Therefore, the evaluation of the Czech PTSA for the Temelín NPP is incomplete.

The safety precautions resulting from the VERLIFE methodology as applied for PTSA to the Temelín NPP are not a valid interpretation of the applicable recommendations of the IAEA Guidelines and do not conform to the requirements of the National and Russian Standards at the time of ETE start-up.

2.3 Issues of further interest, monitoring items about Rules and Regulations

However, since standards, rules, regulations and in particular recommendations in this specific area are still under development, and analyses, as well as the tools used also, the consequences for sound application of PTS mitigation are to steadily observe development and selection of the appropriate tools as well as verification of the current and perspective status of the plant. In order to be able to evaluate the new VERLIFE methodology and its application to the Temelín NPP with respect to international practice it is necessary that the information contained in the document will be made available.

During former workshops, it was indicated that in case of unexpectedly faster embrittlement the utility would consider an annealing of the RPV to reduce the RPV-steel brittleness, the result from neutron embrittlement.

The specialists recommend to consider monitoring the continuous updating of the RPVI assurance and to consider obtaining information on RPV annealing in case it is planned.

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[25] nk=1, $\Delta T=0$ [K]

[26] nk=2 or $\sqrt{2}$, $\Delta T=10$ [K]

3 PRESSURIZED THERMAL SHOCK ANALYSIS (PTSA) STATE-OF-THE-ART AND SPECIFIC CZECH APPROACH

Areas of Monitoring

No	VLI/VLI group description
2	**DESIGN & MANUFACTURING**
2.1	**TECHNICAL BASIS FOR PTSA AND REACTOR PRESSURE VESSEL INTEGRITY**
2.1.1	**General Issues related to Pressurised Thermal Shock Analysis**
1	What is the status documented of the plant specific PTSA? Which parts have been completed, which parts are still pending or ongoing?
2	How have the PTSA analyses influenced the development of the Temelín EOPs (if at all)? What changes to the EOPs have been necessary as a result of the PTSA?
3	What was the extent of peer review performed in conjunction with the PTSA? What organization(s) or individuals were responsible for performing a peer review? What aspects of the PTSA were not included within the peer review? What were the results and conclusions of the peer review? What recommendations for re-analysis or additional analyses were made in the peer review results?
4	In which respect do the SAMGs reflect PTSA related accident management provisions? Were there changes required to the originally used procedures and what was the outcome?
5	Would you describe the way plant specific PTSA results have influenced the development of Symptom Based EOPs as well as SAMGs?
2.1.2	**RPVI assumptions and PTS scenarios**
1	How many accident sequences have been analysed for scenarios with and without operator actions? What are the actions considered and in which accident management measures do these scenarios result?
2	What are operator interventions that are part of the Temelín EOPs? Which of the operator interventions that could influence the progression of a PTS transient (either positively or negatively) were not considered in the PTSA and for what reasons were they not considered?
3	What approaches did the PTSA use to identify the potential for extraneous operator interventions (i.e., actions not contained in the Temelín EOPs), which could influence the progression of a PTS transient (either positively or negatively)? What extraneous operator interventions were identified as a result and how were these interventions reflected in the PTSA?
4	For which PTS scenarios did you analyse with and without operator intervention? In what Temelín EOPs arise operator interventions as modelled for each case, where intervention was considered?
5	Which scenarios have been analysed with regard to success criteria – PTS avoidance and/or mitigation? (e.g. timing and rate of ECCS injection during the transient)
6	What are potential plant upgrades in relation to PTS and RPVI identified as the result of PTSA and accident analyses?
2.1.3	**Modelling aspects of the PTSA**
1	What was the basis for the selection of computer codes to be used for PTSA?
2	Which computer codes were used for the PTSA, in which version were they qualified for WWER 1000 simulations? Please identify according to the specific area of application?
3	For which use were additional models and correlations adopted and/or developed? What were the criteria applied for selection, verification, application and evaluation performed for these specific items?

No	VLI/VLI group description
2	**DESIGN & MANUFACTURING**
4	By what means was each of these computer codes validated for application to the Temelín design? In what reports is the validation documented of each employed code used in the Temelín PTSA?
5	Which organisations (Plant operator, supplier(s), TSOs) were involved in the PTSA (external subcontractors and/or plant staff)? What organization's quality assurance (QA) program governed the overall PTSA? Please explain the work set-up and the tasks associated.
6	For the Regulatory Authority, what organisations have been called upon or are intended to be called upon to review the PTSA for Temelín? What is the scope and extent of the accomplished or planned regulatory review of the PTSA for Temelín?
7	The PTSA, to what extent is it based on best estimate assumptions. In each case where best estimate assumptions are employed in the PTSA, by what means was the assumption validated for its applicability to Temelín? What decision criteria were employed to ensure that in each case the assumption employed represents a best estimate for Temelín?
8	Have benchmarks been conducted for the verification of models and data used in PTSA? Were those benchmarks selected by their applicability to WWER 1000 plants? What was the outcome of these QA measures?
9	Have the models used in the PTSA been verified by application in standard problem exercises, pre-test and/or post-test comparison with experiments, code-to-code comparison tests, sensitivity studies, and comparison with the results obtained for other similar plants? Which scenarios were subject to such comparison? What were the conclusions drawn from such comparisons?
10	Have modelling assumptions and/or parameters been subjected to sensitivity/uncertainty analyses? Which parameters were subject to these analyses? Which accident scenarios were selected for these investigations?
11	Were all criteria/assumptions about specific phenomena identified and clearly defined for the following issues: (a) Determination of sequences of interest and relevance, (b) Performance of thermal hydraulic analysis, (c) Performance of engineering mixing calculations, (d) Calculation of temperature and stress fields, (e) Deterministic fracture mechanics calculations, (f) Verification of the sequence of the engineering approach, (g) Comparison of the thermal hydraulic analysis results, (h) Verification of the mixing behaviour, (i) Verification of temperature and stress fields, (j) Fracture mechanics simulation.
2.1.4	**Adequacy and completeness of documentation of the RPVI and PTSA**
1	Are plant specific data all compiled into a single document/file ("Database of the Analysis")? Is there a comprehensive description on how plant data were converted into a code input deck for PTSA ("PTSA Handbook")?
2	How are the plant-specific data used in the RPVI and PTSA documented? How has this data been validated and validation been added to the documentation? How has the data documentation been archived?
3	Has the selection of optional code/model parameters been properly documented (including justification)?
4	How have the input data for individual PTSA accident scenarios been archived?
5	Are the results of PTSA accident analysis adequately documented? Can the results be properly linked with the specific input deck? Is there a comprehensive description of the PTSA accident scenario and the code version used in the simulation?

No	VLI/VLI group description
2	**DESIGN & MANUFACTURING**
6	Have the selected RPVI management strategies been formally documented?
2.2	**PTS ANALYSIS REQUIREMENTS, CONDUCT AND RESULTS**
2.2.1	**Accident scenario selection in relation to the RPV vulnerability**
1	Do the selected PTS/RPVI strategies reflect the current international knowledge and practices?
2	Are there any specific generic strategies that have not been considered in the plant-specific PTS/RPVI strategies? If so, what is the justification for such decisions?
3	Are the objectives and criteria clearly defined for each of the strategies considered in the plant-specific PTS/RPVI?
4	What are the qualitative and quantitative goals of the RPVI management strategies implemented at Temelín?
2.2.2	**Plant behaviour during accident sequences prior and during PTS events (Asymmetric cool-down behaviour specifics included)**
1	Has the influence of support system failures been evaluated (cooling water, power supply, etc.) in relation to PTS avoidance and mitigation requirements?
2	Have potential design modifications of the existing plant equipment been identified in relation to RPVI?
3	Have feasible design changes of equipment been considered/implemented?
2.2.3	**Adequacy and completeness of the review of plant capabilities as designed**
1	Which of the WWER 1000 specific events denoted in this list have been analysed? (according to the IAEA guidelines) Candidate Transients: 1. Spectrum of postulated piping breaks 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. Feed-water piping break. How does the list of PTS initiating events, which were considered for Temelín compare with the list in IAEA-EBP-WWER-08, Appendix IV? Was there one single master input deck used as the basis for event specific modifications?
2	Have all needs for the use of equipment and related input been determined for the related PTSA simulation? Which input data are based on generic assumptions?
3	Was the entire external input data required (e.g. proprietary designers' and manufacturer's data) made available for the development of the PTSA?
4	To what extent does the Temelín PTSA rely on non-WWER calculations and insights, on non-Temelín specific WWER-1000 calculations and insights? Insofar as it does, how were these simulations and insights qualified to ensure their applicability to ETE?
5	Has the effectiveness of the RPVI management strategies been proven by specific accident analyses, simulator tests etc.?

No	VLI/VLI group description
2	**DESIGN & MANUFACTURING**
6	Do the selected PTS/RPVI strategies cover all relevant functions? Do they include protection of RCS integrity and minimization of radioactivity release?
2.2.4	**Development status of Emergency Operation Procedures (EOPs) related to PTS/RPVI**
1	Have the scenarios contributing significantly to PTS/RPVI risk been identified?
2	Have all the EOP-related symptoms been properly identified? Which parameters are used?
3	Have recovery actions for DBAs been specified and verified?
4	Is information needed to detect level and trend of severity available to the operators?
5	Have the conditions for operator involvement been clearly defined?
6	Have the exit conditions and further steps been defined?
7	What was the extent of the EOPs validation?
8	By what means has it been verified that the PTS avoidance and mitigation procedures and guidelines represent technically correct interpretations of high-level strategies, and that they are capable of achieving their objectives?
9	By what means were the accident scenarios selected which were used for validation and verification of the PTS/RPVI avoidance and mitigation procedures and guidelines, and how was it ensured that the full range of strategies and actions were examined in the course of this validation and verification?
2.2.5	**Tools for PTSA: Thermal hydraulic codes, mixing codes, structure dynamics codes and fracture mechanics methodologies**
1	Which tools have been used for TH simulation (including tools to provide for results analyses, conditioning of results for transfer to other codes, (remapping etc.) and requirements, options/ systems used, to diagnose the solution stability)?
2	Which tools have been used for heat transfer and stress transient simulation (including tools to provide for results analyses, conditioning of results for transfer to other codes, (remapping etc.) and requirements, options/systems used, to diagnose the solution stability)?
3	What methodology has been used for SIF calculations? Have analytical approaches been applied as approved by the Russian general designer?
4	Which postulated defect configurations have been used?
5	Has credit been taken of the WPS (warm pre-stressing) effect? Are there any restrictions on WPS or do the CZ regulations permit to take full credit of WPS?
6	Were coupled codes applied to the neutron kinetics – thermal hydraulic simulations in order to include also – to the extent possible – process feedback?
7	Is there a description of the validation matrix elements for the individual codes, which have undergone validation procedures?
2.2.6	**Dosimetry, neutron spectra/fluence monitoring methodology**
1	What are the most important features of neutron spectra/fluence monitoring programs with respect to the embrittlement monitoring?
2	Has the implemented neutron spectra/fluence monitoring methodology been changed with respect to changes of requirements originating from design changes?
3	What are the uncertainties assumed and/or determined for fluences as evaluated? What were the results in comparison with the absolute – not extrapolated – fluence calculations performed?

No	VLI/VLI group description
2	**DESIGN & MANUFACTURING**
2.3	**PTS ANALYSIS RESULTS IMPLEMENTATION AND VERIFICATION**
2.3.1	**ETE-PTSA results implementation and verification: General aspects**
1	What are the most remarkable PTSA results? What are the most critical accident scenarios with respect to the maximum allowable critical temperature?
2	Have the PTSA results been qualified? Which major qualification procedures/steps were applied to the results?
3	Have PTS avoidance and mitigation measures validation exercises been conducted? If not, what arrangements are planned?
4	Have feasible design changes of equipment been considered/implemented?
5	Was the validation exercise properly designed in order to verify the completeness and adequacy of the PTS operational guidelines? Which accident scenarios were selected for the validation exercise?
6	Is the documentation of the PTS exercise comprehensive (covering the preparation, conduct, results, insights, and conclusions) and in which way is it used as the basis for updating during operation?
7	Have all the lessons from the exercise been properly analysed and used to propose improvements of the guidelines?
8	Did the selection of accident scenarios for validation exercise allow for testing relevant parts of the PTSA findings and roles of different users?
9	What administrative arrangements have been introduced at the plant to control the process of PTSA implementation, verification and QA?
2.3.2	**Internal and external reviews**
1	Have internal and external reviews been conducted of PTSA development and PTS mitigation implementation? What mechanisms and administrative arrangements have been in place to ensure effective feedback from these reviews? Have the recommendations been implemented?
2	What mechanisms and administrative arrangements have been in place to identify potential shortcomings of PTS mitigation and RPVI assurance?
2.3.3	**Management provisions for systematic revision of the PTSA**
1	Are appropriate arrangements/procedures in place at the plant to review the PTS provisions in the future (well-defined and formalized program for conducting reviews at regular intervals)?
2	Is there a formalized program to capture new insights, changes in technology, and modifications of the plant?
2.3.4	**Management provisions for RPVI, PTS and qualifications of the staff**
1	Have staffing/qualification requirements been identified and documented?
2	Are the staffing/qualification requirements complete, i.e. do they contain considerations about the knowledge required for PTS/RPVI?
3	Have appropriate administrative procedures been developed and implemented relating to qualification?
2.3.5	**EOPs and SAMGs: Required Control Room procedures and guidelines for RPVI/PCSI**
1	Do all EOPs and SAMGs contain those provisions intended to preserve RPVI, which have been determined mandatory for the safe conduct of plant operation under adverse conditions? Is there an exemplary procedure available, including the related training program and scheduling?

No	VLI/VLI group description
2	**DESIGN & MANUFACTURING**
2.3.6	**Westinghouse concept**
1	Which provisions were made, that the use of the Westinghouse concept is compatible with National regulations?
2	
3	Which features of the Westinghouse concept make sure that it remains conservative with respect to asymmetric cooling conditions in every respect?
2.3.7	**Compatibility with Czech regulations**
1	Which provisions were made, that the use of the Westinghouse concept is compatible with National regulations?
2	In which way do the Czech regulations call for the application of up to date technology use in simulation and evaluation to be applied in safety relevant verification processes e.g. for CFD codes application?

3.1 PTSA procedure in general

3.1.1 Description of the issue – fundamentals

Principally the demonstration of reactor pressure vessel integrity can be performed in deterministic or probabilistic manner.

Probabilistic analysis considering statistical distributions of the important parameters, such as crack size and density, material characteristics (reference temperature for ductile-brittle transition, etc.), using Monte-Carlo techniques to define the conditional probability of failure for a given transient; multiplying the conditional probability by the probability of occurrence of this transient, and finally summing up the results for all PTS transients is yielding a global failure probability for the PTS case which has to be below a specified value.

Deterministic analyses of the behaviour of postulated cracks in the RPV material during the assumed PTS event, taking into account possible material degradations due to the neutron irradiation. The deterministic demonstration of RPVI performed by a PTS analysis is based on the simulation of the plant behaviour during a selected accident transient and its influence on the thermal hydraulics of the coolant medium in the RPV, the mixing processes in the downcomer fluid, the heat transfer to the RPV wall, the fracture mechanical behaviour of postulated cracks under the resulting stress fields. The calculated load path for a specific accident transient and a specific crack is finally compared with the material state (actual and up to end-of-life) of the RPV steel in order to determine the safety margin throughout operational lifetime.

According to the IAEA Guidelines [IAEA 1997] *"the purpose of the PTS analysis is to provide a reasonably bounding plant specific demonstration of the RPV integrity by using realistic modelling methods for the individual elements of the analysis with conservative assumptions, initial and boundary conditions and appropriate safety factors in the assessment of the results."*

The pressurised thermal shock analysis consists of the following steps:

- Selection of the initiating events
- Thermal hydraulic calculations
- Mixing calculations
- Stress analysis
- Fracture mechanics

According to the IAEA Guidelines, the selection of transients for deterministic analysis can be based on engineering judgement using the design basis accident analysis approach combined with the operational experience accumulated at WWER plants. Another possibility is the selection of transients based on the probabilistic event tree methodology identifying those specific transient scenarios, which would contribute most significantly to the total PTS risk.

General thermal hydraulic and mixing calculations of the cooling-down process during the accident transient give the following parameters as a function of time during the overcooling event (as inputs for the wall temperature and stress calculations):

- Downcomer temperature field
- Coolant-to-wall heat transfer coefficients in the downcomer
- Primary circuit pressure.

Temperature and stress field have to be calculated for all selected PTS sequences as well as cold over-pressurization regimes.

The stress analysis is bound to calculate the stresses due to internal pressure, temperature gradients, and residual stresses (for both cladding and welds including the beneficial effect of the first hydro test if deemed useful). Plasticity effects also should be considered. Stress fields should be calculated for different time steps, which should be selected in a way that peak stresses caused by the transient can be described up to the steady state condition.

The IAEA Guidelines state that simplified fracture mechanics calculations based on formulas such as given in the Russian Code can be used in cases when linear elastic fracture mechanics can be applied for the whole RPV wall thickness including cladding. In the case of complex stress loading especially in the region of elastic-plastic stress state, detailed fracture mechanics calculations using finite elements method (FEM) should be used.

General PTSA requirements and regulations

In order to cope with the threat of a pressurized thermal shock event the National Regulatory authorities introduced into their National Codes defined regulations and requirements for the demonstration of reactor pressure vessel integrity under normal and faulty operation.

<u>Russian Federation</u>

The rules of PNAEG-G-7-002-86 ("Calculations Standard for Strength of Equipment and Pipes of Nuclear Power Units") were applied for NPP Temelín construction, except for PTSA.

For the PTSA, the designer performed the selection of transients and defect sizes for the deterministic analysis. A set of semi-elliptical surface cracks with aspect ratio 2:3 and a relative crack depth up to ¼ of the wall thickness is postulated. The initiation of cracks is not allowed. The cladding is taken into account in thermal and stress analysis, but disregarded for the fracture mechanics. The radiation effects are considered by specified formulas.

For probabilistic analyses, the statistically processed manufacturer's data on the initial flaw size distribution is used for the pre-service PTSA. The probability of vessel failure in case of PTS has to be less than 10^{-7} per year.

<u>United States</u>

In the United States a generic probabilistic analysis was performed for PWR (pressurized water reactors) with the aim to define a simple "screening criterion" that specifies the limiting values for the reference ductile-brittle transition temperature: as long as it is demonstrated that this screening criterion is met during the service life, no PTS analysis has to be performed.

The so-called "screening criterion" defines a maximum acceptable RT_{NDT} (reference temperature for nil ductility temperature, definition see 4.1):

$RT_{PTS} \leq 132$ *[°C] for plates, forgings, axial welds*

$RT_{PTS} \leq 149$ *[°C] for circumferential welds*

where $RT_{PTS} = RT_{NDT}^{unirradiated} + M + \Delta RT_{PTS}$

M................. margin to be added to cover uncertainties in the value of initial RT_{NDT}, Cu and Ni contents, fluence, and in the calculation procedure (M = 36,6 [K] for welds, 26,6 [K] for base metal, in case of generic values for the unirradiated reference temperature, and 18,9 [K] for the base metal in case of measured values.

ΔRT_{PTS}......... mean value of the adjusted reference temperature, calculated according [Reg.Guide 1.99, rev.2]

Germany

The German Code KTA 3201.2 may be applied, in practice more sophisticated analyses are performed. The PTS analysis to be performed is a deterministic fracture mechanical analysis using realistic and worst case scenarios in order to demonstrate the sensitivity to different parameters. The flaw size to be postulated is to times the reliably detectable flaw size with an aspect ratio of 1:3. For realistic transients the absence of crack initiation has to be demonstrated. For worst case transients crack initiation is accepted if crack arrest is demonstrated to occur at a final crack depth smaller than ¾ of the wall thickness.

France

The French Code RCC-M contains in Appendix ZG the general rules, but there is no easily applicable criterion to determine the acceptability of the risk of vessel failure in case of PTS. The PTS analysis is performed in a deterministic way, assuming pessimistic hypotheses the selection of transients and flaw sized is not strictly regulated. A generic study of the risk of brittle fracture for 900 [MWe] RPVs in the beltline region concluded that the limit for safe operation may be conservatively defined as a value of 80 [°C] for the difference of the reference temperature RT_{NDT} for the inner surface materials and the ECCS injection temperature (RT_{NDT} –TIS < 80 [°C]) [EPRI 1990].

IAEA

The IAEA has published guidelines for PTS analysis of WWER reactors [IAEA 1997]. The guidelines provide advice on the individual elements of the PTS analysis, such as acceptance criteria, analysis methods, computer codes, and assumptions to be used as well as on quality assurance.

"The objective of the guidelines is to establish a set of recommendations for RPV PTS analysis, considering related recommendations of the IAEA NUSS Codes, Standards and Guides. The recommendations of these guidelines are based on state-of-the-art practices, operational experience and results of research and development effort in Member States."

"The PTS analysis outlined in the guidelines covers transients and accidents to be considered in the reactor design according to the IAEA Safety Guide 50-SG-D11. The purpose of the PTS analysis is to provide a reasonably bounding plant specific demonstration of the RPV integrity by using realistic modelling methods for the individual elements of the analysis but with conservative assumptions, initial and boundary conditions and appropriate safety factors in the assessment of the results. Deterministic approach is used in the guidelines by analysis of limiting transients from each group of events. Limiting in this sense is understood as limiting from the point of view of RPV integrity."

3.1.2 State-of-the-art requirements and regulations

Requirements for lifetime evaluation of WWER reactor pressure vessels and internals during their operation [SÚJB 1998]

After the political changes in the 1990ies, the National authorities SÚJB (SONS) initiated the preparation of regulatory requirements for lifetime evaluation of reactor components including aspects of integrity and degradation processes covering the following issues:

1. Requirements and criteria for lifetime evaluation of WWER RPVs – general requirements
2. Requirements and criteria for lifetime evaluation of WWER RPV internals – general requirements
3. Approach and principal procedure for evaluation of the RPV resistance against non-ductile failure
4. Procedure for determination of radiation field in RPV and its internals
5. General requirements
6. Requirements for computational determination
7. Requirements for experimental determination
8. Procedure and criteria for an evaluation of acceptability of defects found during in-service inspections.
9. Procedure and requirements for an evaluation of the effect of pressurized thermal shock on RPV behaviour.
10. Procedure and requirements for mechanical testing of surveillance specimens
11. Procedure for an application of results from surveillance specimens programme testing to RPV lifetime evaluation
12. Procedure for a fatigue damage evaluation
13. Procedure for a corrosion and corrosion-mechanical damage evaluation
14. Procedure and requirements for instrumented hardness measurement of components in operation
15. Requirements for repair welding procedures of RPVs and for evaluation of their effect on RPV lifetime

This document has the character of guidelines, is not a mandatory regulation, but the fulfilment of these requirements will simplify the acceptance of the RPVI assessment by the Regulatory Authority SÚJB.

3.1.3 Current plant status

Requirements and regulations as applied for ETE

Workshop 2004 presentations: M. Brumovský, J. Žd'arek: Reactor pressure vessel integrity (RPVI) assurance approach; V. Pištora: Comparison of IAEA, Russian and VERLIFE methodologies for PTS assessment, M. Svab, Z. Mokersky: SÚJB comment on current legislation basis aspects to RPV PTS.

Nuclear power plants in the Czech Republic were built under the agreement between the former Czechoslovakia and USSR in the context of mutual co-operation. Within that agreement the Soviet design, production standards and rules were used for NPP realisation. The construction of NPP Temelín was therefore started within the frame of these Russian Standards [BRUMOVSKY 2001]:

- Rules for design and safe operation of components and piping of NPPs (PNAE G-7-008-89).
- Standards for strength calculation of reactor components, steam generators, vessels and piping of nuclear power plants, test and research reactors and appliances (1973).
- Regulations for inspections of welded joints in nuclear power plants, experimental and test reactors (PK 1514-72).

ASI Code

The Czech Association of Mechanical Engineers (ASI) is working on Codes for WWER reactor components. The format of the SÚJB requirements is supposed to be consistent with the Code that is prepared by ASI.

The planned Code will have five sections:

Section I	Welding and brazing of components and piping of WWER type NPPs
Section II	Characteristics of materials for components and piping of WWER type NPPs
Section III	Strength assessment of components and piping of WWER type NPPs
Section IV	Evaluation of residual lifetime of components and piping of WWER type NPPs
Section V	Material testing procedures and evaluation

According to [BRUMOVSKY 2001] the main problem for the ASI Code is the fact that only Soviet type material were allowed for use in WWER type reactors according to the Russian Codes. Some of these materials are no more produced in the National Czech factories.

Section III was prepared for the design state of calculations but is also applicable for the evaluation of components during operation.

Section IV will be applied for the main components of the primary circuit, incl. the reactor pressure vessel. This section has the following structure:

1. Introduction
2. Basic principles, nomenclature and definitions
3. General requirements for examination and inspections
4. Requirements for pressure tests
5. Principle procedure for lifetime evaluation of components and piping

Appendix A	Procedure for determination of radiation field in RPV and internals
Appendix B	Requirements for surveillance specimens program and its testing
Appendix C	Procedure for application of results from surveillance specimens testing to RPV material degradation
Appendix D	Assessment of fatigue damage in components and Piping
Appendix E	Assessment of corrosion and corrosion-mechanical damage in component and piping
Appendix F	Requirements for a choice and evaluation of pressurized thermal shock regimes
Appendix G	Evaluation of non-destructive examination results
Appendix H	Evaluation of acceptability of defects

Statements made during the Workshop in Prague (May 24/25th, 2004) confirmed that the Code is not yet finalized and enacted.

VERLIFE Methodology (2004) [ANNEX B]

The “unified procedure for lifetime assessment of components and piping in WWER NPPs, VERLIFE” was developed within the project of the 5th Framework of the EU (EC 5FP; October 1st, 2001 to September 30th, 2003). VERLIFE is the methodology of the structural part of the PTS assessment for Temelín NPP and Dukovany NPP[27], report DITI 301/267, UJV Řež, a.s., 2004. The VERLIFE methodology was approved by SÚJB in the beginning of May 2004. The other Regulatory Bodies of the participant countries of the project have not yet accepted the VERLIFE for RPV lifetime and integrity evaluation [BRUMOVSKY 2004b], [PISTORA 2004a], [SVAB 2004a].

The procedure defined in VERLIFE *“can be used for evaluation of residual lifetime of components and piping of NPP with WWER type reactors during their operation”*, can be used *“for elaboration of Periodic Safety Reports to demonstrate operational safety and reliability of components and piping during reactor operation”* and can be used *“for a definition of conditions for further reactor operation within or beyond the component or piping design lifetime/ licence validity.”* The VERLIFE document has 12 appendices:

1. Structure of the report assessing residual lifetime of equipment
2. Procedure for determination of neutron fluence in reactor pressure vessel
3. Assessment of degradation of properties of materials
4. Determination of values of stress intensity factor KI
5. Determination of reference/design fracture toughness curve including “Master Curve” approach
6. Requirements for pressurized thermal shock (PTS) selection and thermal hydraulic calculations
7. Residual lifetime of the equipment damaged by fatigue due to operating loading
8. General recommendation for piping and components temperature measurement
9. Assessment for corrosion-mechanical damage of materials
10. Schematisation of flaws
11. Tables of allowable sizes of indications found during in-service inspections
12. Evaluation of defect allowance in components

Due to the fact that the substance of the VERLIFE methodology, including the 12 appendices and information about the state of completion is not available, the Austrian Experts can only rely on the presentations of the Workshop [BRUMOVSKY 2004b], [PISTORA 2004a] and other comparable publications regarding VERLIFE [BRUMOVSKY 2003a, BRUMOVSKY 2003b, BRUMOVSKY 2003c, BRUMOVSKY 2003d, BRUMOVSKY 2003e].

The following changes concerning PTSA (RPVI under emergency conditions) with respect to the Russian Code and the IAEA Guidelines were identified in the presentation [PISTORA 2004a]:

- The main difference to the Russian (originally Soviet) rules is the application of the “Master Curve” approach, traditional transition temperatures based on Charpy impact test data are allowed as a secondary alternative.
- The fracture toughness curve $K_{Ic}(T)$ differs from the Russian Code definitions, but is identical to the IAEA fracture toughness curve; the Master Curve fracture toughness curve $K_{Ic}(0,05)$ is also more or less similar to the IAEA curve, except that there is no limitation for high values. Both curves are more conservative than the Russian version

[27] Metodika pevnosti cásti hodnocení tlakove-teplotních soku (PTS) pro JE Temelín a GJE Dukovany

- The maximum crack depth is restricted to $^{1}/_{10}$ of the total wall thickness, while the older Russian Code and the IAEA Guidelines require an analysis up to ¼ wall thickness (without cladding)[28].
- For the postulated cracks VERLIFE requires only semi-elliptical cracks, while the Russian Code requires elliptical cracks[29], IAEA requires both, semi-elliptical and elliptical cracks and [SÚJB 1998] semi-elliptical surface cracks in case the cladding is not 100% tested or defectuous and elliptical underclad cracks in case of a tested and defect-free cladding.
- The aspect ratio requirements in VERLIFE is $^{a}/_{c}$ = 0,3 and 0,7 (identical to IAEA; Russian Code: 0,333)
- Fatigue cracks are not taken into account (contrary to the Russian Code and IAEA)
- Residual stresses due to the cladding are included by definition of the stress free temperature equal to the operation temperature (Russian Code: 390 [MPa] in the cladding, below the cladding dependent on heat treatment temperature and time)
- Residual stresses in the welds: $\sigma = 60.\cos(2\pi x/S)$ [MPa] for axial and circumferential stresses (Russian Code: axial stress $\sigma = \sigma om.\cos(2\pi x/S)$ [MPa], circumferential stress: $\sigma = \sigma om$, dependent on tempering temperature and time; both constant through wall thickness – for ETE σom is about [100 MPa].
- The biaxiality of shallow cracks is not taken into account (in the Russian Code the biaxiality is considered)
- The adjustment of stress intensity factor calculations to the crack front length is not considered for the T_k approach, but is considered within the Master Curve approach (similar to the Russian methodology)
- In the VERLIFE application, no safety factors for the loading KI (emergency conditions) as used (the Russian Code uses $(1{,}1K_{IP} + K_{IS})$[30]). The IAEA Guidelines require for the case that only cracks be considered that be smaller than ¼ of the wall thickness (as VERLIFE does) the use of a safety factor of √2 for the stresses or a safety factor of 2 for the crack size.
- The WPS (warm pre-stressing effect) is applied with 90% of global maximum of KI (as in the new version of the Russian Code; the IAEA Guidelines allow the application of 80% of the global maximum of K_I[31])
- Crack arrest can be applied (cannot be applied according to the Russian requirements; IAEA states: “It should be noted that other complementary approaches could be used provided they are properly justified and validated, such as crack arrest approach for postulated accidents. In such cases, specific acceptance criteria may need to be defined.”)

3.1.4 Evaluation

NPP Temelín construction was started under former Soviet auspices according to the Soviet design and manufacture regulations. Even during the late construction phase under former Czechoslovakian and later Czech Republic authorities the Russian Code, regulations were the only legal regulatory base.

The Czech ASI Code (see above) is still in the state of elaboration. Nevertheless, SÚJB claims that the current legislation is based on:

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28 In [SÚJB 1998] the postulated crack depths of in case of qualified NDT programmes may be reduced from ¼ to $^{1}/_{8}$ of the wall thickness.

29 Provided that cladding NDT is performed and no flaws are detected above the allowable size.

30 K_{IP}: primary stresses, K_{IS}: secondary stresses

31 In [SÚJB 1998] the application of WPS was not included

- Section IV (Residual lifetime assessment of WWER nuclear power plants components and piping)
- The instructions and recommendations for lifetime assessment of WWER RPV and reactor internals during NPP operation (see above)
- IAEA Guidelines on PTSA for WWER nuclear power plants [IAEA 1997]

According to the SÚJB representative the new VERLIFE project was accepted by Association of Mechanical Engineers of the Czech Republic (ASI) as a document for assessment of components and piping in WWER NPPs.

Austrian Experts asked during the Workshop 2004 whether the VERLIFE Methodology will have the status of a mandatory rule; the Czech answer was, that the application of the VERLIFE methodology is not mandatory, but in case this methodology is applied for RPVI demonstration SÚJB will accept the results.

It is obvious that there are no legalized National Codes/Standards that define regulatory PTSA requirements applicable for the NPP Temelín in the state of design, construction and licensing. Therefore, the valid and thus applicable Code requirements for PTSA are the Russian Standards as valid during the state of design and construction in accordance with the presentation of the SÚJB representative.

The IAEA Guidelines [IAEA 1997] prepared for the structural assessment of WWER reactors are still basis for the RPVI assessment, according to SÚJB even part of the current legislation.

Therefore, it seems to be necessary to evaluate the substantial reductions of conservatism in the VERLIFE application with respect to the IAEA Guidelines. Within the presentation [PISTORA 2004], the comparison was performed with the Russian Code – considering the new methodology (from 2000), not the Russian Code that was valid during design and construction.

The three mandatory standards at the time of ETE start-up and cited in the POSAR ([PNAE-G7-002-086], [SÚJB 1998] and [IAEA 1997]) require for the demonstration of the RPV structural integrity under PTS conditions a postulated crack size of ¼ wall thickness.

- [PNAE-G7-002-086]: paragraph 5.8.5.2: "For the calculation of the stress intensity factor (SIF) the postulated crack is a semi-elliptical surface crack with a crack depth of a = 0,25 s (s: wall thickness) and the aspect ratio $^a/_c$ = 2:3." No exceptions are allowed. Additional calculations for small cracks are required, as they might eventually become unstable before the $^s/_4$ crack due to the effect of temperature and pressure gradients.
- [SÚJB 1998]: paragraph 1.3.9.3: SIFs have to be calculated with semi-elliptical cracks with a crack depth ranging from a = 0 to a_{postul} with a_{postul} = 0,25 s, $^a/_c = {}^2/_3$ and $^a/_c = {}^1/_5$ in case of unknown fracture toughness of the cladding – in case of a defect-free cladding with known fracture toughness the SIFs calculations have to be performed for elliptical cracks with a_{postul} = 0,125 s[32], $^a/_c = {}^2/_3$ and $^a/_c = {}^1/_5$. Paragraph 1.3.9.11: In case of a qualified NDE program, a_{postul} can be reduced to $^s/_8$.
- [IAEA 1997]: paragraph 6.3: "The postulated defect should be defined in the following way:
  - *For uncladded vessels, the postulated defect is a surface semi-elliptical crack with depth of ¼ of the RPV wall thickness and with an aspect ratio $^a/_c$ in the range of 0.3 to 0.7.*
  - *For cladded vessels, cladding integrity of which is verified by redundant non-destructive testing and its mechanical properties are known the postulated defects are undercladding elliptical as well as semi-elliptical cracks with depth up to ¼ of the RPV wall thickness, and with aspect ratio $^a/_c$ resp. $2^a/_c$ in the range of 0.3 to 0.7.*

[32] Since the elliptical crack has a depth of 2a, the effective depth of apostul is identical to that of the semi-elliptical crack with a depth of a

- *For cladded vessels, where limited or no information on cladding exists, the postulated defect is a surface through cladding semi-elliptical crack with depth up to ¼ wall thickness and with aspect ratio $^{a}/_{c}$ in the range of 0.3 to 0.7.*

Usually the analyses of cracks with aspect ratio of 0.3 and 0.7 are sufficient." "Defect sizes smaller than ¼ wall thickness could be used for the RPV integrity assessment under PTS loading of <u>plants under operation</u> if it is possible to demonstrate the required non-destructive testing reliability and if permitted by the national regulatory requirements."

"A parametric analysis to identify the conservative value of the postulated defect aspect ratio $^{a}/_{c}$ in the range of 0.1 to 1.0 should be considered."

According to the Czech Experts (at the Workshops in Prague and Řež) the VERLIFE methodology is also to be applied in case of pre-service PTSA with a = $(s_N + s)/10$, i.e. $^{1}/_{10}$ of the total wall thickness (cladding + ferritic RPV wall). The total wall thickness of the WWER-1000 RPV is for the cylindrical part 200,5 [mm]. Thus, the new methodology reduces the crack size to be postulated from 50 [mm] to 20 [mm], which is a safety reduction by the factor 2,5. This sudden change of the safety philosophy is obviously triggered by the fact that the demonstration of RPVI under PTS conditions is not possible with postulated crack depths of ¼ of the wall thickness

Consequently based on these new criteria one has to ask about the results of the PTSA and which changes to the PSA results must be expected.

Safety factors

For defects smaller than ¼ of the wall thickness in the IAEA Guidelines the following safety factors for postulated accidents are given:

Safety factors	**A**	**B**	**Application of the safety factors**
n_k [1]	1	√2	(uncertainties with respect to loading functions)
n_a [1]	2	1	(uncertainties with respect to postulated crack size)
ΔT [K]	10	10	(uncertainties with respect to fracture toughness curve)

"Out of the two sets of safety factors[33] given, the set yielding less favourable results should be used in the assessment."

No safety factors are defined or recommended in the VERLIFE requirements.

In [SÚJB 1998], the safety factors for the calculation of the stress intensity factors were already eliminated (n_k = 1,0.δT = +0 [K]).

The argumentation of the Czech Experts during the Workshop was that the IAEA Guidelines are being reviewed with the option of reducing the conservatism.

Actually, the draft of revision 2 states, "the safety factors n_k and ΔT should not be applied simultaneously, the procedure of this safety factor application should be expressed in the following form:

$$K_I(T_k^a) \leq \min \{K_{Ic}(T)/n_k \,;\, K_{Ic}\,(T+\Delta T)\}".$$

This draft of IAEA Guidelines, revision 2 recommends the use of n_k = 1,1 in case of postulated defect smaller than $^{S}/_{8}$ (S: wall thickness).

[33] Sets A or B (inserted by the author)

The issue of safety factors within the methodology of Western countries is not comparable because of the very different methodologies: In Germany general fracture mechanical concepts are defined as being applicable, but in practice more sophisticated analyses are being performed; the flaw size to be considered is equal to two times the reliably detectable flaw size, assuming $^{a}/_{c} = ^{1}/_{3}$. In France, the safety margins are dependent on a combination of flaw sizes and transient categories. These procedures are not comparable with the discussed VERLIFE methodology. The uncertainties of the reference temperature determination are not covered by the regulations in Germany and France; the U.S. regulations require the use of a safety margin to cover the uncertainties of the experimental method for the determination of the initial RT_{NDT} and the uncertainties of the determination of ΔTRT_{NDT} (15,5 [K] for welds and 9,5 [K]). Other National Codes do not provide rules for the use of safety margins to consider the uncertainties.

Taking into account all the uncertainties with respect to the basis of embrittlement predictions (indications on the non-conservatism of the Russian code specification, small entity of the available plant-specific data, scatter band of the experimental data from irradiated samples), the requirements set by the IAEA Guidelines to use a safety factor are fundamental with respect to RPVI.

Warm pre-stressing (WPS)[34]

With respect to the applicability of the WPS effect the IAEA Guidelines [IAEA 1997] state:

"In the assessment, warm pre-stressing could be credited for loads below 0,8 peak stress intensity factor in the continuously decreasing crack loading path, utilizing the assumption, demonstrated by large scale testing, that crack initiation does not occur in the decreasing crack loading path."

The Russian Standards [PNAE-G-7-002-86] and [SÚJB 1998] do not allow to take credit of the WPS effect.

The VERLIFE methodology uses the 90% criterion. The argumentation of the Czech Experts for the deviation from the IAEA requirements is that the new Russian methodology does also use the 90% criterion. At the time of design and manufacture of the Temelín RPV, the Russian Code did not allow taking credit of the WPS effect.

The application of the WPS effect is allowed in the U.S., but in reality not applied because of the involved uncertainties. In Germany, it was not allowed to take credit of the WPS effect before 1996, but was applied for the demonstration of RPVI in the NPPs Stade and Obrigheim. In France WPS is not even mentioned in the Code regulations. In Finland WPS is applied only for large LOCA.

It has to be stated that the use of the 90% criterion instead of the IAEA Guideline 80% criterion causes a large reduction of residual safety margins.

Crack size/shape

For the postulated flaws the VERLIFE, methodology defines semi-elliptical cracks for integrity demonstration. The flaws are oriented in axial as well as circumferential direction, underclad location (provided that both integrity of the cladding has proved by NDT and mechanical properties are known), with crack depth a = (SN+S)/10, $^{a}/_{c}$ = 0,3 and 0,7. One mandatory prerequisite is qualified NDT; in case of a crack detected during ISI the acceptability check on this crack needs to be performed by fracture mechanics analysis using a factor 2 for determining the crack model size; this means that a crack size of at least 10 [mm] must be re-

[34] The WPS effect will be discussed in detail in the chapter on the comparison on the comparison of the load path with the fracture toughness curve

liably detectable, in order that the postulated crack size may be reduced to $^{1}/_{10}$ wall thickness, which in the case of ETE with SN+S = 200,5 [mm] is 20 [mm]. Fatigue crack growth is not being accounted for in this context.

In Germany, the details of the fracture mechanical analysis are not defined by KTA – but usually semi-elliptical cracks are assumed. The French Code RCC-M, Appendix ZG, does not strictly regulate the flaw sizes. Based on a consensus between the contractor, the utility and the safety authorities combinations of defect sizes and transient categories according to the probability of their occurrence are considered. These reference defects for the vessel beltline include elliptical cracks within the cladding, elliptical underclad cracks and semi-elliptical surface cracks. Due to these basic differences in the methodologies, it does not seem to be appropriate to compare the Western Code regulations with the Czech VERLIFE approach with respect to the flaw issue.

In comparison with the Russian Code and the IAEA Guidelines the VERLIFE approach is less conservative, because [IAEA 1997] calls for semi-elliptical and elliptical crack shapes, the Russian Code assumes elliptical flaws, provided that no flaws exceeding the allowable sizes in the cladding were detected by NDE methods.

The national Czech Guidelines [SÚJB 1998] require the postulation of semi-elliptical surface cracks in case of unknown fracture toughness of the cladding or alternatively elliptical underclad cracks in case of known fracture toughness of the cladding and the demonstration of a crack-free cladding.

The Czech Experts [PISTORA 2004a] performed a broad quantitative comparison of VERLIFE with the Russian methodology showing that the restriction to semi-elliptical cracks has a strong influence on the maximum allowable critical temperature of brittleness. This result has not been taken into account in the Czech PTS analyses, and it must be assumed that this was done in accordance with the Supervisory Body[35].

The Czech Experts have performed a comparison of the resulting maximum allowable critical temperature of brittleness for VERLIFE and the Russian methodology, but not with IAEA Guideline application.

Consequently, no quantitative comparison has been performed with respect to the strongest reduction of conservatism by

- The change from the 80% criterion according [IAEA 1997] concerning the application of WPS to the 90% criterion in VERLIFE, and
- The elimination of the safety factors (n_k=1 and ΔT=0).
- A preliminary estimating consideration of these two conservatism reducing approaches applied to the presented results of T_k^a values [PISTORA 2004] shows that this new approach will allow considerable additional service life time for the WWER-1000 (and presumably for the whole WWER projects). Lasting operation of these plants will very probably is possible only when the internationally (by the IAEA) defined safety factors are eliminated and consequently the conservatism defined by the IAEA for these plants is strongly reduced (see also 3.6 and 3.7).

[35] Workshop in Řež, remark by Mr. Pištora: The considerably more critical elliptical crack was introduced in the Russian Code in order to compensate for the lack of conservativity in other places. However in Temelín in several instances procedures are more conservative than those in the Russian Code. Because of this it was decided not to pile up conservatisms more than necesssary in the Temelin case.

3.2 Accident scenario selection

3.2.1 Description of the issue – fundamentals

Due to the complex variety of interactions between different components and additional operator actions, it is difficult to assess all possible pressurised thermal shock inducing events. In accordance with the IAEA Guidelines, the PTS initiating events may be grouped with respect to the frequency of occurrence. Events with higher probability need to be treated with requirements that are more stringent. The IAEA Guidelines [IAEA 1997] define two categories:

Anticipated transients (AT)

Anticipated transients are defined as relatively frequent deviations (frequency of occurrence higher than 10^{-2} per reactor year) from normal operating conditions, which are caused by malfunction of a component or operator error. These transients should not have safety related consequences to RPV integrity, which would prevent the plant operation to be continued.

Postulated accidents (PAs)

Postulated accidents are defined as such rare deviations from normal operation which are not expected to occur (less than 10^{-2} per reactor year on the average) but are considered in the original design or in the design of plant upgrading or in the course of plant safety reassessment. For these events, not only plant operation is interrupted, also immediate resumption of operation may not be possible.

A complete analysis of the RPV response to deviations from normal operation has to consider all credible events that might threaten the RPV integrity. It is rather difficult to decide about the selection of a limiting or bounding accident transient. Therefore, the IAEA has compiled a list of initiating accidents that should be considered for the selection of a bounding case assessment for PWRs:

- Loss of coolant accidents (LOCA)
- Stuck open pressuriser safety or relief valve
- Primary to secondary leakage accidents
- Large secondary leaks
- Inadvertent actuation of high pressure injection or make-up systems
- Accidents resulting in cooling of the RPV from outside

Loss of coolant accidents may occur with different sizes of both cold and hot leg breaks which are characterized by rapid cooldown. Especially those scenarios that lead to flow stagnation causing faster cooldown rates and cold plumes in the downcomer might be critical. Attention should be given to breaks of auxiliary pipes connected to primary system (for instance the pressuriser surge line). The IAEA states the *“Cold repressurization of the reactor vessel is usually prohibited in principle, but since the isolation of breaks gets high priority in the operating procedures of the WWER reactors, the possibility of isolating the leak and resulting re-pressurization has to be considered.”*

3.2.2 State-of-the-art requirements and regulations

The selection of accident scenarios for a PTS analysis is highly dependent on the specific reactor type. In the case of Temelín, it is therefore appropriate to rely on the IAEA Guidelines for WWER-reactors [IAEA 1997]:

The Appendix IV of the IAEA Guidelines [IAEA 1997] contains a special list of initiating events recommended for consideration with WWER-1000 NPPs:

1. Spectrum of postulated piping breaks 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:
5. SG tube rupture
6. Primary collector leaks up to cover lift-up.
7. Inadvertent opening of one check or isolation valve separating reactor coolant boundary and low-pressure part of the system.
8. Inadvertent actuation of ECCS during power operation.
9. Chemical and volume control system malfunction that increases reactor coolant inventory.
10. Inadvertent opening of one steam generator safety or relief valve or turbine bypass valve.
11. Spectrum of steam system piping break inside and outside of containment.
12. Feedwater piping break.

3.2.3 Current plant status

Workshop presentations: P. Kral, Selection of Scenarios for PTS Analyses and TH Methodology

According to [KRAL 2004] the following accident scenarios were selected for the Temelín PTS analysis:

A	Loss-of-coolant accidents (LOCA)
A1.	Small break LOCA with break size of ND32 mm in the cold leg (2 scenarios with max/min ECCS configuration, zero reactor power)
A2.	Small break LOCA with break size of ND60 mm in the cold leg (2 scenarios with max/min ECCS, zero reactor power)
A3.	Medium-break LOCA with break size of ND125 mm in the cold leg (2 scenarios with max/min ECCS, zero reactor power)
A4.	Medium-break LOCA with rupture of the PRZ SV pipe ND210 mm (4 scenarios with max/min ECCS and full/zero reactor power)
A5.	Large-break LOCA with break size of ND300 mm in the cold leg (2 scenarios with max/min ECCS, full reactor power)
A6.	Large-break LOCA with break size of ND300 mm in the hot leg (2 scenarios with max/min ECCS, full reactor power)
A7.	Double-ended guillotine rupture 2xND850 in the cold leg (1 scenario with min ECCS and full reactor power)
A8.	Double-ended guillotine rupture 2xND850 in the hot leg (2 scenarios: max ECCS+HZP, min ECCS+N100%)
B	**Increase of heat removal by secondary circuit**
B1.	Main steam line break (MSLB) close to SG (2 scenarios with full/zero reactor power)
B2.	Main steam line break upstream of MSIV (2 scenarios with full/zero reactor power)
B3.	Main steam line break between MSIV and MSH (1 scenario with zero reactor power)

B4.	Main steam header rupture (MSHR) (2 scenarios with full/zero reactor power)
B5.	Main steam line break downstream of MSH (1 scenario with zero reactor power)
B6.	Inadvertent opening of steam dump to atmosphere (SDA) (1 scenario with zero reactor power)
B7.	Inadvertent opening of steam dump to atmosphere (SDA) (1 scenario with zero reactor power)
C	**Leaks from primary to secondary side of steam generator**
C1.	Rupture of 1 SG tube (1 scenario with max ECCS and zero reactor power)
C2.	Rupture of 3 SG tube
C3.	SG primary collector cover lift-up (2 scenarios with max/min ECCS, zero reactor power)
D	**Other initiating events**
D1.	Inadvertent opening of PRZ SV and its reclosure (2 scenarios with max/min ECCS, zero reactor power)
D2.	Inadvertent actuation of high pressure injection system (HPIS) (2 scenarios with various initial regime of the unit)
D3.	Make-up system malfunction leading to increase of RCS inventory (1 scenario)
D4.	Analysis of feed & bleed (EOP procedure FR-H.1) (1 scenario)
D5.	Interface LOCA (1 scenarios with max ECCS and zero reactor power)
D6.	Inadvertent start of HA injection due to operating personnel mistake (1 scenario)

3.2.4 Evaluation

For some of these important accident transients the PTSA has been performed by the Czech Experts (A7, A8, B1, B2, B3, D1, D2 …). The task is still not completed, esp. the analyses have to be performed for the group of mid-size breaks (A2 to A6) that might be the most important ones with respect to critical effects.

3.3 Thermal-hydraulic calculations

3.3.1 Description of the issue – fundamentals

The general thermal hydraulic calculations give the temperature field in the RPV as a function of time during the overcooling event for the wall temperature and stress calculations.

The overcooling transients that threaten the RPV integrity are usually very complex so that it is often not possible to define in advance conservative or limiting conditions for all system parameters. Engineering judgement might not be sufficient to decide whether an accident under consideration is a PTS event or will result together with other consequences in a PTS event.

Therefore, thermal hydraulic analysis might be necessary for the selection of initiating accidents, events and scenarios, which can be identified as limiting cases.

A basic requirement is the adequacy of the physical model being used to represent plant behaviour. This might depend on the accident being evaluated.

The cool-down processes should be simulated up to parameters where the primary coolant circuit can be considered stabilized. In most emergency core cooling cases, this means that the temperature of the primary circuit reaches the temperature of the water the emergency core cooling system's storage tanks.

The cool-down rate has to be determined by taking into consideration various aspects as follows:

As long as the natural circulation is maintained, a uniform cool-down of the entire primary circuit can be assumed.

If flow stagnation occurs in the primary system, the cooling process has to be investigated for significantly smaller control volumes; since colder plumes will develop extending into the downcomer and as a consequence, temperature and heat transfer coefficient distributions tend to be non-uniform and therefore asymmetric.

There are specific assumptions for flow stagnation cases:

- In case of a LOCA compensated by ECCS, when the reactor coolant pumps are tripped and the decay heat level is very low, the flow stagnates when loop flow rate is outweighed by the core cooling injection rate;
- For a LOCA non-compensated by emergency core coolant delivery the onset of the flow stagnates when steam enters hot legs.

The models should include an accurate presentation of the pertinent part of the primary and secondary systems. Particular attention should be given to the modelling of control systems.

The thermal hydraulic models have to be capable of predicting single and two-phase flow behaviour and critical flow as required, the models should be capable of predicting the plant behaviour for selected accident scenarios such as LOCAs, steam line breaks, primary-to-secondary leakage accidents, and different overcooling transients.

The computer codes used for thermal-hydraulic calculations have to be validated.

3.3.2 State-of-the-art requirements and regulations

National codes do not prescribe the use of specific codes. Therefore, it seems to be appropriate to rely on the IAEA Guidelines [IAEA 1997].

With the two objectives of thermal hydraulic analysis, to support the transient selection process and to produce necessary input data for structural analyses the following aims are sought:

According to common practice, thermal hydraulic calculations should give the following parameters as a function of time during the overcooling event for the wall temperature and stress calculations:

- Downcomer temperature field;
- Coolant-to-wall heat transfer coefficients in the downcomer;
- Primary circuit pressure.

Transients' selection and in advance definition of conservative or limiting conditions for all system parameters is usually very complex. The selection should enable to decide, whether an accident scenario should be considered a PTS event or consequences can lead to a PTS event, which may potentially threaten RPV integrity. As a result, those initiating events and scenarios should be identified, which are limiting cases within the considered group of e-vents. The development of overcooling scenarios derived from the transient selection is a key activity in the thermal hydraulics assessment, which uses the same thermal-hydraulic system codes as in the accident analysis.

There is a trend to use coupled codes also in the PTSA with 3-D or quasi 3-D capabilities applied particularly to the nozzle-downcomer regions. Also needed for the mixing and heat transfer calculations in particular, the calculation method employed for the thermal-hydraulic analysis already takes into account the need for the description of 3-D fluid behaviour over time during the transient.

It has been recognized, that the simulation must be capable to map the normal operation systems' control, main feedwater and make-up system, the pressurizer for calculating pressure and eventual repressurisation of the primary circuit. The structural analysis can be performed with simplified pressure and temperature curves only for very fast cooldown like in the LB-LOCA. Thermal stratification of high-pressure injection water in the cold leg could be a problem for correct simulation of the downcomer temperature and heat transfer distribution.

In addition to those, a series of specific conditions should be defined, derived from degraded normal upset conditions by supplementary failures, which would increase the severity of the transients. For that purpose, the different states of the reactor (operating, hot, intermediate and cold standby) are to be considered to define the envelope of transients. The corresponding frequency evaluation allows classifying the conditions in higher ranks than the initiating ones. This part of the study should provide complementary transients relevant to RPVI verification.

3.3.3 Current plant status

Workshop presentations:

Král, P.: Selection of Scenarios for PTS Analyses and TH Methodology;

Král, P., Muhlbauer, P., Malačka, M.: Overview of TH Analyses Results for PTS.

The Czech party presented an Overview of the status of thermo-hydraulic analyses then and about further work. So far, over 35 thermal system analyses and 14 mixing analyses have been performed in the frame of PTS study for the NPP Temelín. Remaining PTS scenarios will be analysed until the end of 2004, when the Temelín PTS Project will be terminated.

Four categories of PTS transients were to be evaluated:

#	Category	# of TH analyses	# of MIX analyses	Status
1	Main steam line break (MSLB) or false opening of dump valve – system thermal-hydraulics analyses (one event was analyzed using both codes RELAP5 and ATHLET) and mixing analyses	13	4	Completed
2	Primary-to-secondary leaks (PRISE) – systems thermal-hydraulics analyses (plus numerous sensitivity calculations) and mixing analyses	4	2	Completed
3	Loss-of-coolant accidents (LOCA) – systems thermal-hydraulics analyses	17	5	Completed
4 a	Inadvertent opening of PRZ PORV			In progress
4 b	Other events: erroneous initiation of HPIS or make-up system injection, Feed & Bleed etc.			To be accomplished

Some comments on the selection of scenarios:

The meaning of “minimum ECCS configuration” presumptive boundary conditions is, that the minimum number of the ECCS systems is available (e.g. $^{1}/_{3}$ HHPIS + $^{1}/_{3}$ HPIS + $^{1}/_{3}$ LPIS + $^{2}/_{4}$ HA + no PCMS), however, it doesn’t mean that the lowest pump characteristics are applied, much to the contrary, always the maximum pump delivery height versus mass flow characteristics are used in combination with minimum safety injection (SI) water temperature etc.

The Czech party selected the following cases one from each group of Initiating Events (IEs) for the PTS Workshop presentations at the Workshop:

1 Main steam line break at SG1 (“SLB1B”)

2 Break of 3 SG tubes (“3SGT”)

3 Large break LOCA in hot leg (“H850 max”)

4 Inadvertent opening of PRZ SV with re-closure (“PSV43b”)

For these transients only minimum details concerning boundary conditions have been presented and the parameters presented representative for the results did not sufficiently characterise the work accomplished.

<u>Description of the scenario „H850max“:</u>

Double-ended guillotine break of the loop-4 hot leg (2xND850) from Hot Zero Power
(HZP ≡ 1% N_{nom} at BOL):

- Minimum reactor coolant flow (80 000 [m^3/h]), at maximum primary temperature and pressure Maximum secondary pressure (7,4 [MPa]), AFWP, SDC Assumed loss-of-offsite power (LOOP) at 0 [s]
- Assumed „maximum ECCS“ configuration for fast total cooldown
- Single failure ≡ failure of spray pump (to minimize temp. of SI)

Boundary conditions selection does not observe strictly conservative assumptions: The worst PTS conditions would result from maximum coolant flow from the ECCS.

<u>Corresponding variant and sensitivity calculations associated with the LBLOCA subcategory:</u>

Break location (CL vs. HL)

Break discharge coefficients (0,8x, 1,0x, 1,2x)

Minimum/maximum ECCS

With/without Loss-Of-Offsite Power

Nodalization in 1D/2D of reactor downcomer

Timing of main PTS related events in the transient H850max

Transient time [s]	Events – Actions
0	Break (DEGB in HL4, 2xD850 mm)
0	LOOP + trip of all RCPs + reactor SCRAM
0,1	Safety-Signal, TQ-request (from continuous overpressure exceeding+30 [kPa])
2	First occurrence of reverse flow in the broken loop
4,6	Start of 4/4 HA injection
16	Start of injection of 3/3 HPIS into CL1, 3,4
21	Start of injection of 3/3 LPIS into RCS (CL1, HL1, UP, DC)
21÷27	First flow reversal in intact loops
64	End of HA injection
200	HPIS suction switchover of from tanks to sump
245	Minimum of reactor inlet temperature (21,5 [°C] from CL1)
3600	End of calculation

Example of results no.4 – Inadvertent opening of PRZ SV (“PSV43b”)

Description of the scenario „PSV43b“:

Transient time [s]	Events – Actions
0	Inadvertent opening of PRZ SV1 from HZP (1% N_{nom}) at BOL • Minimum core coolant flow (80 000 [m^3/h]), maximum CL temperature, maximum prim. pressure • Maximum secondary pressure (7,4 [MPa]), AFWP, SDC • Assumed loss-of-offsite power (LOOP) at • Assumed „maximum ECCS“ configuration for fast total cooldown • Single failure ≡ failure of spray pump (to minimize temperature of SI fluid)
1700	PRZ SV re-closure (the most adverse time for this scenario)
1800	Operator stop of HPIS (according to FR-P.1, step-6)

The selected fraction for the residual power (1% N_{nom}) is too high, in other calculations 0,1% N_{nom} is selected. From the results is clear, that HHIS is assumed not available. This does not comply with a conservative approach.

Corresponding calculations (in „PSV“ subcategory – not finished yet):

+ Full Reactor power/hot zero power
+ Minimum/maximum ECCS
+ Various time of PRZ SV re-closure
+ Operator actions etc.

Timing of main events in „PSV43b“ (with respect to PTS)

Transient time [s]	Events – Actions
0	Inadvertent opening of PRZ SV1
0	LOOP + trip of all RCPs + reactor SCRAM
72	Rupture of PRT safety discs (= outflow to containment)
133	S-SIGNAL, TQ-NEEDED (from low HL subcooling)
134	Injection of HPSI into CL1,3,4
210 ÷ 215	First flow reversal in primary loops
420	Re-switch of HPIS suction from tanks to sump
515	1st minimum of R inlet temperature (17,8 [°C] from CL3)
1040	2nd minimum of R inlet temperature (18,0 [°C] from CL1)
1700	Reclosure of PRZ SV1 (inadvertent)
1780	Reaching of HPIS shut-off head (11,7 [MPa]) and zero SI flow
1800	Operator stop HPSI (EOP: FR-P.1, step-6)
2900	Maximum primary pressure (12,7 [MPa])
7200	End of calculation

Inadvertent opening of pressuriser safety valve System TH analyses:
(First subgroup of „other events“ group)

TH mixing + structural analyses completed for the following 2 scenarios:

Transient time [s]	Events – Actions
	PSV43 – inadvertent opening of PRZ SV1:
0	Starting at hot zero power, Maximum ECCS, No PRZ SV re-closure,
1800	Operator stop of HPIS
	PSV43b – inadvertent opening of PRZ SV1:
0	Starting at hot zero power Maximum ECCS
1700	PRZ SV re-closure
1800	Operator stop of HPIS

Structural Analysis:

Conservative „enveloping“ structural analysis of PSV43 scenario with pressure equal to shut of head of HPIS pump = 11,7 [MPa] (during the entire time interval analysed), simulating in this way possible PRZ SV re-closure in arbitrary time. Analysis was performed until HPIS stop at 1800 [s].

From the detailed presentation of the results follows in all the cases analysed that HHIS is not considered available.

3.3.4 Evaluation

There are objections with respect to the details of the accident transients. For instance for the accident scenario of inadvertent opening of the pressuriser safety valve the loss of off-site-power was not assumed, although in this case the re-pressurisation after a random re-closure of the valve would be significantly higher. The IAEA Guidelines required for that scenario the additional consideration of the total loss of offsite-power.

The selection of base cases compares well to those described in the [IAEA 1997] recommendations and some of these cases have also been taken up in in-depth analyses conducted by TSOs treating other WWER-1000 installations, which provide indications about the sequence of events intended to serve PTS avoidance during the transients [IVANOVA 2000], [GROUDEV 2000]. Along these lines an effort was made to also gain insight into the selection of PTS sequence aggravated conditions to the RPV resulting from thermal-hydraulic rapid cooldown events. These exercises, even conducted for a generic WWER-1000, confirmed some of the considerations taken up by the Czech side.

It must be mentioned that deviations of conservative approaches have been identified along the thermal-hydraulic transient sequences. These include, as it became clear from the limited information provided:

- Maximum coolant flow from the ECCS was not assumed
- Lowest pump characteristics were used instead of maximum delivery of cold coolant
- Minimum safety injection water temperature was not applied in all cases
- HHIS is assumed not available for additional cooldown velocity
- Secondary cooling is not clear with respect to contribution to the transients cooling.

In general, a quite extensive investigation has been performed in terms of transient numbers. The selection of the most severe cooldown processes to be expected should therefore have achieved a solid background. Knowing that the entire process of verification of RPVI/PTS is still going on, one can assume that for further consolidation additional information will also be used, as it is available.

3.4 Mixing calculations/Heat transfer

3.4.1 Description of the issue – fundamentals

The application of the thermal hydraulic systems analyses code and the subsequent fluid-fluid mixing code is supposed to provide for the time dependent temperature distribution field simulation results in the downcomer for the structure analyses, the heat transfer coefficient simulation time dependent results fields and evolution of the primary pressure during selected transients and accidents.

The quasi-3D methods applied in mixing codes currently in use are based on engineering models or on the regional mixing model. They allow an accurate calculation of the extent of the thermal stratification in correspondence with the overall system response.

A promising tool for the proper prediction of the (more complex) turbulent mixing of fluids of different temperatures and velocities within a complex geometry might be a combined three dimensional general purpose fluid dynamic analysis code applying a finite element or finite difference technique. However, these methodologies are still in the process of development and validation.

3.4.2 State-of-the-art requirements and regulations

For the determination of the exact temperature, fields in the downcomer of the RPVs the internationally enforced codes do not contain any distinct regulations besides the requirement that such calculations have to be performed subsequent to the general thermal-hydraulic calculations.

The IAEA Guidelines [IAEA 1997] state:

"The calculation methods, employed for the PTS analysis, should be qualified for this purpose."

"The integrity assessment of nuclear components relies strongly on the validity of the computer codes used for structural and fracture mechanics analysis. The development of the codes is guided by the principle that the phenomena of interest can be described to a sufficient level of accuracy. A predictive structural code is typically a compilation of individual mechanisms (e.g. material behaviour, deformation, heat conduction, crack behaviour), which are combined into analytical models. Usually the choice of the model to describe the mechanisms is based on a variety of assumptions. Therefore, the validation of a code must take into account the procedures to model the individual mechanisms including simplifying assumptions."[36]

The IAEA Guidelines [IAEA 1997] contain exemplary mixing calculations, based on the computer code REMIX/NEWMIX for the NPP Loviisa, Finland in Appendix 7. This program system code is based on engineering knowledge and has to be adjusted to the specific plant to be treated.

The results of these mixing calculations include temperature transients and the changes with time of the heat transfer coefficient at the centre line and outside of the cold plume. The global geometry of the cold plume (width v/s height), the temperature distribution and the heat transfer coefficients distribution in all nodalisation points within the plume are then described by known analytical approaches.

This methodology can be considered a state-of-the-art procedure for WWER reactors. The verification of different computer codes for the calculation of mixing processes has been discussed based on large-scale experimental data. The computer codes REMIX/NEWMIX have demonstrated applicability for WWER reactors.

The development of CFD codes based on 3D representation of at least extended areas of flow, be it single or multiphase flow, is for the time being for the most demanding applications not in such an advanced stage, that results from such calculations could be considered a stand alone endorsement for acceptance criteria fulfilment. Namely, the following NRC policy assertion for the CFD code application in the mixing-heat transfer calculations reflects rather well the current status:

"Single-phase computational fluid dynamics (CFD) techniques have been in use for several years by the NRC to study special problems, e.g., thermal mixing in the downcomer during the study of pressurized thermal shock. These single-phase CFD techniques provide detailed flow-field information but are limited to conditions of single-phase flow. Pipes and vessels are routinely broken down into as many as 10,000-100,000 computational cells for CFD analysis. Two-phase CFD techniques are needed to enable detailed CFD solutions to be obtained under conditions of two-phase flow as experienced in nuclear reactor vessels and piping during accident conditions."

[36] Assessing the status of validation of any computer code is difficult because there has not been a formal consensus on what constitutes a validated code. In some countries (USA, UK) procedures to certify code capabilities have been developed.

And in another context:

"There are some issues about single-phase flow CFD that will have to be addressed by the NRC before CFD methods can be certified. The two most significant issues are the choice of the computational grid and the turbulence modelling. The main concerns with the computational grid have to do with geometric fidelity and numerical convergence. Fortunately, there are many tools available for mesh generation, and the commercial codes can use many different types of meshes. For example, it is now possible to use tetrahedral elements that can be adjusted to any configuration. The question of numerical convergence is resolved by mesh refinement. A simple rule is to double the mesh until the solution stops changing. However, in 3D calculations each doubling increases the number of mesh points by a factor of 8 and it is easy to overload even the most powerful computers.

The selection of valid turbulence models is a little more delicate. Most CFD codes use some version of the k-e model. However, even though this model is a considerable improvement over the mixing length models of the past, the k-e model remains controversial. Perhaps the main issue from a practical point of view is that there is no general k-e model, but there are several variations that work better for certain flow configurations. Therefore, the validation of specific turbulence models for different flow geometries, and the determination of the uncertainties are the key considerations.

The success of single-phase flow CFD has raises the possibility of two-phase flow CFD. However, the presence of deformable interfaces makes this problem much more difficult. In addition, the effect of the interfaces on the turbulence must be considered. Considerable progress has been made on dispersed flows (i.e., droplet and bubbly flows) but much work still remains to be done. The two-fluid model is the most practical approach although the Lagrangian-Eulerian method is also used. Both ways it is necessary to determine the interfacial area concentration, and this is the major source of uncertainties. However, recent transport models for the interfacial area concentration that consider various coalescence and break-up mechanisms are very promising."

Therefore the value added by treating single-phase flow situations this way – as these are applicable for most of the PTS events treated here – could be substantial. Particularly for coalescent cold tongues simulation and the associate heat transfer estimates the CFD code applications are of indisputable value – and will be even more supportive in the future. See also [SIEVERS 2000], [MAZZINI 2003c]

3.4.3 Current plant status

Workshop presentation:

P. Král, V. Pištora: Selection of Scenarios for PTS Analyses and TH Methodology
[KRAL 2004a]

P. Král, P. Mühlbauer, M. Malačka: Overview of TH Analyses Results for PTS [KRAL 2004b]

The [KRAL 2004a] statements with regard to mixing indicate the codes applied and the related SÚJB certification for that purpose:

- Performed for all cases with non-symmetric or non-homogeneous cooldown of reactor downcomer
- The goal: to determine the thermal loading of the RPV (basic output to structural analysis = 2D temperature field at inner surface of RPV wall)
- In cases with ideal mixing in cold legs and no radial stratification in the downcomer ("cold sectors") or two-phase conditions in the downcomer ("cold stripes"), CATHARE code used
- For transients with flow stagnation and thermal stratification in cold legs and predominantly single-phase conditions in the downcomer ("cold plumes"), the REMIX or NEWMIX code modified for NPP Temelín used.

REMIX/NEWMIX is used for the stratification cases, whereas the two-phase flow cases with perfect mixing in the cold legs are treated with the French code CATHARE in a 2-D modelling approach. Phase separation cases have not been touched in this context at all.

In [KRAL 2004b] the following information on the realization of the mixing calculations has been presented:

- In case of ideal mixing in the cold legs and no radial stratification in the downcomer ("cold sectors") or two-phase conditions in the downcomer ("cold stripes") the CATHARE 2 Ver. 1.5 code was used.[37]
- For transients with flow stagnation and thermal stratification in the cold legs and predominantly single-phase conditions in the downcomer ("cold plumes") the REMIX/NEWMIX codes modified for NPP Temelín were used.

The general background and the geometrical model were described during the presentation and example calculations were demonstrated.

3.4.4 Evaluation

For the accident transients within the PTS analyses the mixing calculations were performed comparable to current international practice and in compliance with the calculations for other WWER reactors.

It was recognised by the Experts' team that in some instances benchmarking was applied in order to verify codes capabilities and reconfirm proper simulation conduct.

No specific mixing related conservatisms were mentioned in the presentations. Not even implied conservatisms were explicitly mentioned. The specifically coarse 2-D mesh used with the CATHARE mixing simulation calculations would indicate that stability problems have been overcome by keeping out local effects and avoid including any turbulence effects. Verification against 1-D CATHARE model results is in line with this assumption.

The overall TH generated "boundary conditions" for the mixing process seem to have obtained sufficient care, since evaluation took place with the use of RELAP5 Mod3.2, ATHLET, and for the Containment functional parameters COCOSYS.

The very elaborate discussion of the TH-transients simulations performed was matched with exemplary sets of representations as REMIX graphs from the pool of results:

- Flow stagnation originating examples and
- Thermal stratification originating from single-phase situations in the downcomer.

The Froude number triggered CL stratification criterion conforms to current practices applied. In these cases, flow situations convergent from the various loops in combination would be of interest. Such results have not been explicitly touched.

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[37] (CATHARE 2 Ver.1.5 capabilities:
- Special pseudo 3D module, one of the standard module (continuation of 2D module developed for downcomer modelling)
- Advantages: two-phase flow conditions, downcomer model with all inlet nozzles
- Input parameters from TH calc.: liquid and steam temperatures, liquid and steam velocities, void fractions, pressures, mass flow rates
- Output parameters for fracture mechanics calculation: wall temperatures, liquid temperatures)

The PTS simulation needs are covered only in part by results obtained with CATHARE simulations for “cold tongues” or better “stripes” – since no radial stratification simulation is possible in the 2-D simulation of the downcomer – and two-phase flow would only be applicable for discrete boundaries as they develop. The coalescent, converging type of flow pattern in the downcomer did not turn up in the exemplary mixing calculations results. These results would be needed to gain an insight into the sensitivity considerations for PTS assumptions.

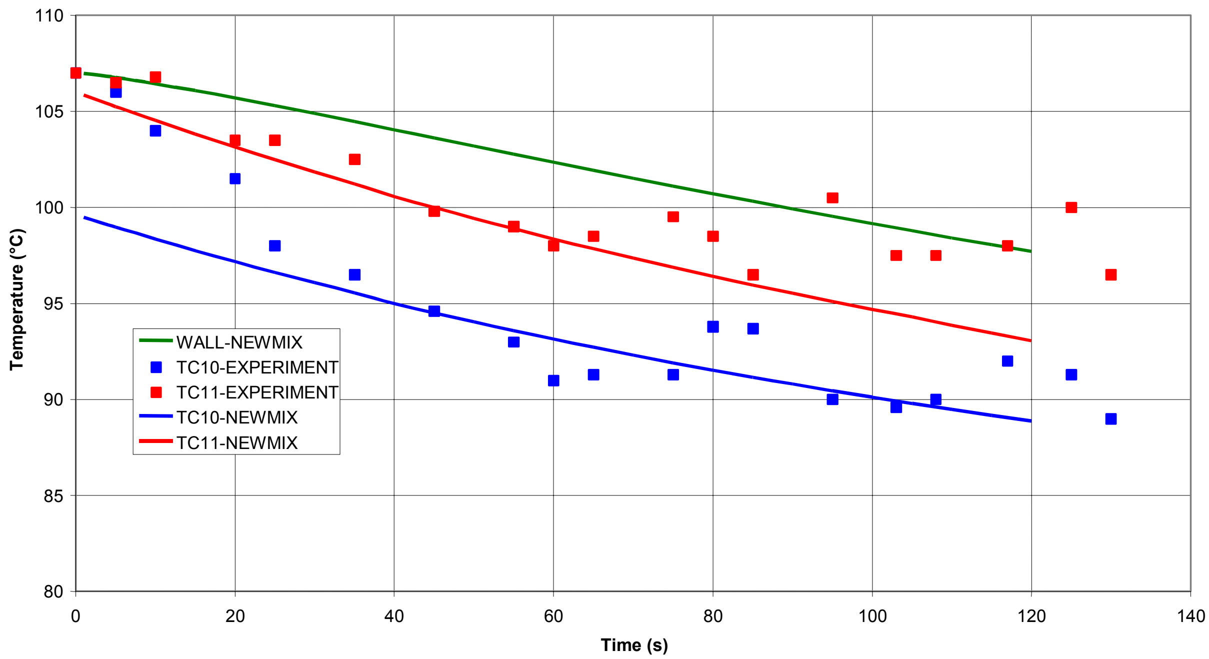

Figure 1: Temperatures of wall and cold plume water in reactor downcomer at locations of reactor pressure vessel welds (positions of thermocouples TC10 a TC11 and results from simulation calculations. [HAUEROVA 2000]

The wrap-up of mixing simulation results would imply also the need for a more detailed introduction of heat transfer and conduction assumptions made, in order to allow for a consecutive appreciation of the merits of the temperature field development simulation along the transients investigated, in the reactor pressure vessel wall base material, cladding, binding zone and weldment regions as well. The heat transfer parameters mapping to the RPV wall in the respective areas and assumptions about the boundary layer behaviour as well as transition simulation models would be of fundamental interest for proper monitoring (see Figure 1). In an exemplary case provided, the location of the thermocouples for the measurement are not identified, however in both cases, be it the fluid or the wall temperature transient, the comparison suggests at least the using up of all credible conservatism (to less than $\Delta T = 5$ [K] – in this case) in the first place, even though the implied SIF maximum might arise at the transient reaching the last test readings shown if not later in the continuation not shown with lower ECS flow rates and thus a bulk fraction of “stagnant” primary coolant swapped into the downcomer [HAUEROVA 2000]. In addition, also the statement, that heat transfer parameters are the result of RELAP and ATHLET simulations raise doubts about the soundness of the approach taken. A discussion of boundary layer heat transfer conditions with and without flashing flow and its simulation options as well as the conduct of the transient would eventually also lead to reconsideration, since the scatter of temperatures even before flow and mixing instabilities dominate the DC energy transfer.

The Czech Experts have provided additional information about heat transfer assumptions intended to endorse the argument of abundant conservatism exceeding comparable PTSA application cases [ANNEX G]. The quality of the arguments appears to be acceptable, however quantification of their significance according to the range of various components of conservatism would have to be determined for appreciation of the credit taken and the extracted boundaries to be respected for the conservative formulation of heat transfer. The review of these results will have to be performed using a set of pilot calculations suitable for comparison. For CFD and 3D applications, the simulation would require ample varieties to be analysed involving considerable resources. In this context the currently accepted state of science and technology has been identified as follows:

The 3-D simulation options currently under development have been neglected in the PTSA process. These options would be an essential asset once a specification of required benchmarks is being added to their application for the use in the plant operation authorization process. This is necessary in order to make sure not only the code is qualified but also the simulation bases, including the heat transfer and mixing.

3.5 Fracture mechanics – SIF calculations

3.5.1 Description of the issue – fundamentals

Based on the temperature field in the RPV wall as result of the mixing calculations the stress field resulting from internal pressure, thermal stresses, residual stresses due to the cladding and in the circumferential welds has to be determined by fracture mechanical methods. Numerical calculations using FEM codes and analytical models are applicable.

The effect of the resulting stress field on a postulated flaw in the RPV wall is described by the stress intensity factor KI. These calculations have to be performed for the bottom of the crack front and at the intersection of the crack with the surface.

3.5.2 State-of-the-art requirements and regulations

National Codes do not prescribe details of physics models or computer codes to be used, but some safety relevant assumptions such as the postulation of defects and the use of safety factors are regulated.

The IAEA Guidelines [IAEA 1997] recommend to validate the applied FEM computer codes for the calculation of the temperature and stress fields or to use computer codes that are widely applied in the international community.

For the calculation of the stress intensity factors both, the application of FEM computer codes or engineering knowledge based analytical approaches may be used. The new Russian code has included analytical approaches based on weight functions.

Postulated defects

In Germany, the details of the fracture mechanics analysis are not defined by KTA – but usually semi-elliptical cracks are assumed. The French Code RCC-M, Appendix ZG, does not strictly regulate the flaw sizes. Based on a consensus between the Contractor, the Utility and the Safety Authority combinations of defect sizes and categories of transients according to the probability of their occurrence are considered. These reference defects for the vessel beltline include elliptical cracks within the cladding, elliptical underclad cracks and semi-elliptical surface cracks.

Since the IAEA Guidelines that were elaborated based on the Russian Code, are part of the Czech current legal basis, the respective requirements should be taken into consideration:

Russian Federation

The older Russian Code [PNAE G-7-002-86] regulations required the assumption of semi-elliptical cracks in the RPV wall (without cladding) with an aspect ratio of 2:3 and crack depths up to ¼ of the RPV wall thickness.

The new Russian Code requires crack depth variations only up to 2a = 0,7s (s=RPV wall thickness). This reduction of conservatism compared to the original Code requirements is very probably a consequence of the fact that the more restrictive safety requirements cannot be met at some of the Russian WWER power plants.

IAEA Guidelines

The IAEA Guidelines [IAEA 1997] requirements for the crack depth variations allow defect sizes below ¼ of the wall thickness only with a permit by the National authority and the use of qualified NDT methods (at least two redundant methods) that provide reliable information on the defect-free state of the cladding.

"For the purpose of the RPV PTS analysis, defects are postulated in the RPV with the objective to demonstrate by fracture mechanics analysis that the acceptance criteria are met for these postulated defects. The postulated defects are surface or subsurface cracks, located in the limiting areas of the vessel. In selection of the limiting areas of the vessel, consideration should be given to the stress level, to the material degradation and to the results of the non-destructive testing. The orientation of the postulated defect should be axial or circumferential depending on the direction of the maximal principal stress.

For cladded vessels, cladding integrity of which is verified by redundant non-destructive testing and its mechanical properties are known, the postulated defects are undercladding elliptical as well as semi-elliptical cracks with depth up to ¼ of the RPV wall thickness, and with aspect ratio $^{a}/_{c}$ resp. $2^{a}/_{c}$ in the range of 0.3 to 0.7.

For cladded vessels, where limited or no information on cladding exists, the postulated defect is surface through cladding semi-elliptical crack with depth up to ¼ of the RPV wall thickness and with aspect ratio $^{a}/_{c}$ in the range of 0.3-0.7.

Usually, the analyses of cracks with aspect ratios of 0.3 and 0.7 are sufficient.

*Defect sizes smaller than ¼ of the wall thickness could be used for the RPV integrity assessment under PTS loading of **plants under operation** if it is possible to demonstrate the required non-destructive testing reliability and if permitted by the national regulatory requirements. In such cases, the shape and the size of the postulated defect should be conservatively assessed with respect to qualified detection and sizing capabilities of NDT ISI used at the given plant. The national standards for NDT and related standards for schematization of detected defects should be taken into account. It is also recommended to apply at least two redundant qualified NDT techniques of different physical principle if small surface cracks are of concern. The size of the postulated defect could be selected with respect to the size of a realistic manufacturing defect probable to exist in the vessel."*

Czech Republic

[SÚJB 1998]: paragraph 1.1.1: *"The true detailed calculation of the residual lifetime is performed with a postulated crack, that is defined as either a semi-elliptical surface crack or as an elliptical under-clad crack with a depth equal to ¼ of the wall thickness of the RPV."*

According to international practice calculations of the temperature and stress fields are usually done assuming elastic-plastic material behaviour of the cladding, the fracture-mechanics calculations (stress intensity factor calculation) is performed for semi-elliptical cracks without taking into account the existence of the cladding. Thus, the behaviour of a semi-elliptical surface crack in a given temperature and stress field is only determined by the fracture-mechanics properties of the reactor steel (base material, weld metal and heat affected zone in the transition region towards the base material).

The following reasons are the basis for this procedure:

- The mitigation contribution effect of the cladding is already taken into account within the temperature and stress field calculations. Due to the immediate contact with the reactor, coolant the cladding is catching the steepest part of the temperature gradient (stress mitigation in comparison with an uncladded RPV) and experiences the lowest temperatures. During a thermal shock event the cladding induces mitigating compressive stresses in the reactor wall due to the higher thermal expansion coefficient. In case of a purely elastic behaviour of the cladding, these stresses are unlimited and directly proportional to the temperature differences. Using a correct assumption describing the elastic-plastic material behaviour of the cladding, these compressive stresses within the cladding are limited by their yield strength.
- No validated knowledge and especially no analytical approaches exist for the case that a crack front ends directly at the interface (ferritic RPV steel –cladding), were an abrupt change of the material characteristics would have to be assumed. Such interface boundary considerations cannot yet be resolved reliably, not even when using FE methods, since they strongly depend on for instance the used approach for the SIF calculation (energy flow method or crack opening approach).
- Taking into account the cladding for the fracture-mechanics calculations would have to be based on profound knowledge of the cladding material characteristics, not only the standard material strength data from tensile testing, but also fracture mechanics properties. Any of the underclad crack models would become obsolete in case the cladding would be defective or fail.
- The cladding has very inhomogeneous properties. Delta ferrite in the cladding is above 60% wt and strong gradients over the cladding thickness were observed. The behaviour of the cladding exposed to neutron irradiation is not to well investigated, the first surveillance samples are currently being irradiated.

With respect to the crack depth [SÚJB 1998], paragraph 1.3.9.3, equation (3.13):

The stress intensity factors have to be calculated for a = 0 to a = a_{postul} with a_{postul} = 0,25 s (s: wall thickness). According to paragraph 1.3.9.11: in case of qualified NDE program a_{postul} may be reduced to $a_{postul}^{qualify}$ = s/8 (equation 3.19).

Safety factors

For defects smaller than ¼ of the wall thickness in the IAEA Guidelines, the following safety factors for postulated accidents are applicable:

Safety factor	A	B	Application of the safety factor
n_k	1	√2	(Uncertainties with respect to loading functions)
n_a	2	1	(Uncertainties with respect to crack size)
∆T [K]	10	10	(Uncertainties with respect to fracture toughness curve)

Out of the two sets of safety factors[38] given, the set yielding less favourable results should be used in the assessment.

No safety factors are defined or recommended in the VERLIFE application.

[38] Set A or B (insert by the author)

In [SÚJB 1998], the safety factors for the calculation of the stress intensity factors were already reduced (n_k = 1,0.δT = + 0 [K]).

During the Workshop, the Czech Experts argued that the IAEA Guidelines are being reviewed with the option of reducing the conservatism.

Actually, in the draft of revision 2 it is stated, *“the safety factors n_k and ∆T should not be applied simultaneously, the procedure of this safety factor application should be expressed in the following form:*

$$K_I(T,a) \leq min \{K_{Ic}(T)/n_k ; K_{Ic} (+\Delta T)\}\text{”.}$$

This draft of the IAEA Guideline, revision 2 recommends the use of n_k = 1,1 in case of a postulated defect smaller than $^S/_8$ (S: wall thickness).

The safety factors within the methodology of Western countries are not comparable because of the very different methodologies:

In Germany general fracture mechanics concepts are defined applicable, but in practice more sophisticated analyses are being performed. The flaw size to be considered is equal to two times the reliably detectable flaw size, assuming $^a/_c$ = ⅓.

In France, the safety margins depend on flaw sizes and transient categories.

Regulations in Germany and France do not cover the uncertainties of the reference temperature determination, whereas the U.S. regulations require the use of a safety margin to cover the uncertainties of the experimental method for the determination of the initial RT_{NDT} and the uncertainties of the determination of ΔT_{RTNDT} (15,5 [K] for welds and 9,5 [K]). Other National Codes do not provide rules for considering the use safety margins to cover uncertainties.

3.5.3 Current plant status

Workshop presentation by A. Kacor: Integrity Models Description

For the PTS analysis at NPP Temelín the FEM computer code SYSTUS (developed and maintained in France, and accepted by SÚJB in 2003) was applied for the temperature and stress fields and for the FEM calculations of the stress intensity factors (G-theta method), i.e. the crack-FE-model was integrated into the mesh representation. For each accident transient 32 3D-FE-models were adopted for the model of one half of the reactor pressure vessel, varying the following parameters:

- Crack depth a = 8, 12, 16, and 20 [mm]
- Aspect ratio $^a/_c$ = 0,3 and 0,7
- Crack position: weld nr. 3 and nr. 4 (according to RPV weld plan)
- Crack orientation: longitudinal and circumferential

Along the crack front the FEM mesh was generated in such a way that in total 21 nodes were available for the stress intensity factor calculation along the semi-elliptical ligament. The calculations were performed considering the elastic-plastic behaviour of the material.

3.5.4 Evaluation

The methodology and calculations of the temperature and stress field using the FEM computer code SYSTUS and assuming elastic-plastic material behaviour were performed apparently in accordance with the current state-of-the-art.

The Czech Experts have chosen unusual procedures for stress intensity factor (SIF) calculations. Using FE-methods for the SIF calculation requires a considerable computational effort. Because of this major computing effort, involved in the integration of the postulated cracks' FE representation into the model, the variation in crack positions and crack geometries calculated is restricted to four different crack depths, two different aspect ratios, two weld positions and two crack orientations. On the other hand, this method allows detailed calculations of the stress intensity factors along the crack front.

The unusual part of the Czech approach is to include the cladding into the SIF calculation (see also paragraph 3.5.2). This requires the assumption that the cladding remains intact with the postulated crack underneath extending only into the ferritic reactor steel [KACOR 2004]. A comparative investigation presented during the workshop in Prague [PISTORA 2004a] included an elliptical underclad crack and a semi-elliptical crack in order to describe the stress intensity factor along the crack front.

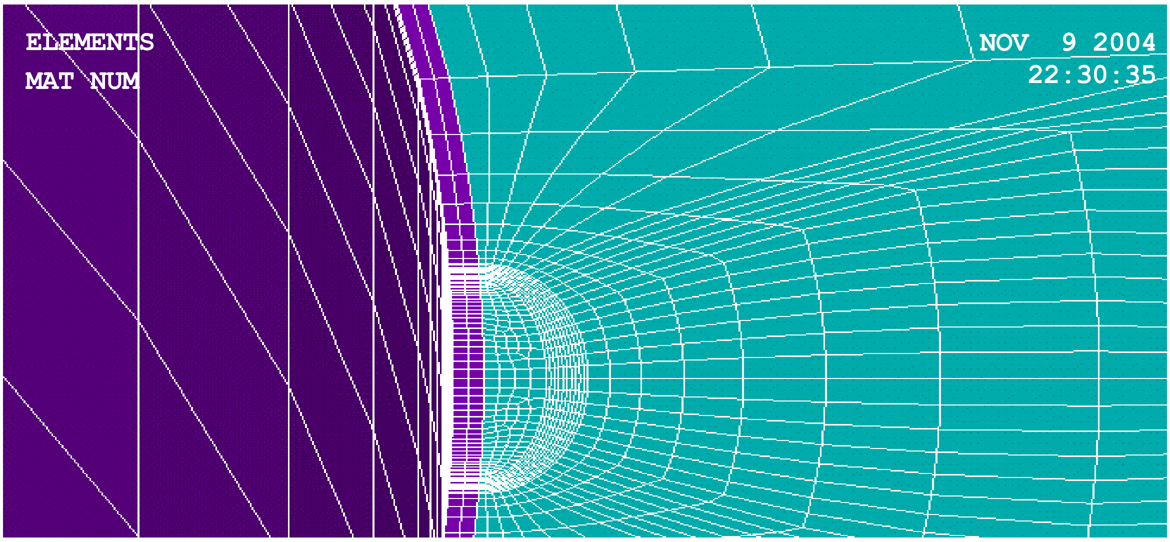

Figure 2: FEM model with Circumferential Semi-elliptical undercladding crack, a=19,25 mm, $^a/_c$ = 0,3, closer view, lower crack half top view.

According to [SÚJB 1998], an elliptical underclad crack representation has to be postulated in case of an intact cladding with known fracture toughness. Semi-elliptical and elliptical crack representations have to be analysed according to [IAEA 1997]

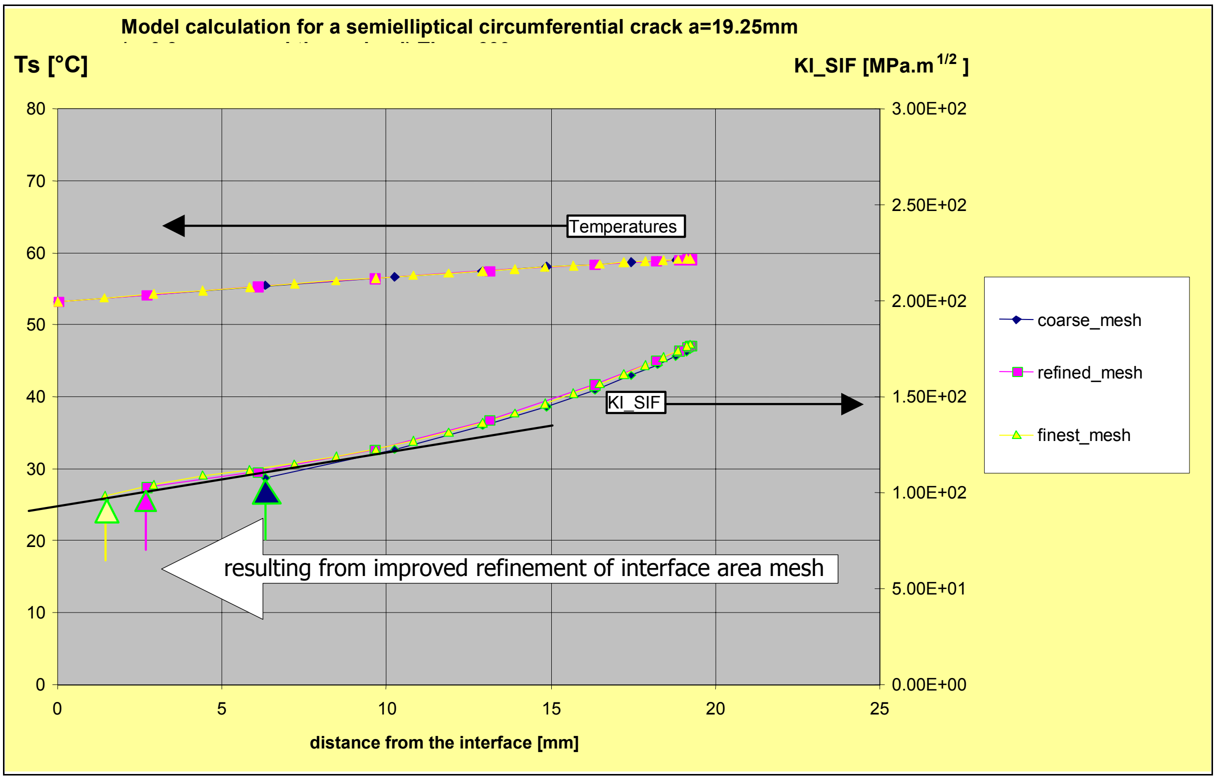

Figure 3: $K_{I\ SIF}$ and temperatures Ts along the crack front, for 3-refinement models

It has been brought to the Austrian Experts' attention in the brief presentation of the VERLIFE methodology application, that semi elliptical cracks are the one and only crack configuration to be considered for the ETE PTSA.

Moreover the resorting to a rather unusual simulation that assumes SIF decreasing locally to SIF = 0, as it has been done in the Czech approach, is contradicted by the results of pilot-case simulations conducted in the monitoring process. Whenever refinement is applied to a mesh inserted for the crack tip at the cladding/base-material interface area the resulting SIF does not exhibit the behaviour assumed in the Czech approach (see Figure 3). This is a clear indication, that the reduction of conservatism is cause by the modelling approach.

The results of the comparative Czech calculations in [PISTORA 2004a], page 23, are commented as follows:

- In case of the semi-elliptical surface crack the stress intensity factor is decreasing significantly towards the interface with the cladding (depending on the accident transient up to a factor of 2). This is supposed to be caused by the defect-free cladding hindering crack opening at the cladding interface. This effect is a reduction of conservatism. Such a stress intensity factor decrease does not appear in case of SIF calculation without cladding. The results of the analytical approach show a rather continuous behaviour of SIF along the crack front. In many cases of the PTSA the crack initiation does not occur in the lowest point of the crack ligament, but in the transition point at the cladding (here is the lowest temperature, the highest temperature gradients and stress gradients and the highest neutron embrittlement, and the stress intensity factor only slightly lower compared to the lowest point). If – based on the assumption of a defect-free intact cladding – the calculations yield a much lower stress intensity factor, the crack initiation at this point does not occur at all or it occurs much later. The resulting T_k^a values are higher than in the more conservative model.

- According to [PISTORA 2004a] an adverse effect appears for the case of an elliptical underclad crack (the ligament touches the interface with the cladding in a single point): The stress intensity factor increases significantly near the transition region to the cladding (by about 60% compared to the average value in lower regions). This effect is not understandable and certainly does need further discussion and analysis. The Czech Experts did not include this effect in their subsequent PTSA since the VERLIFE methodology requires only semi-elliptical surface cracks and the presented results might include uncertainties. In case further studies would confirm the effect of a strong increase of the stress intensity factor near the cladding interface all PTSA results would have to be considered non-conservative.

It must be noted, that significant reductions of conservatism have been introduced step by step into the PTSA:

- *Reduced postulated crack depth under the assumption of qualified NDT:*

[IAEA 1997]	up to smaller	$^s/_4$
[SÚJB 1998]	up to	$^s/_8$
[VERLIFE 2004]	up to	$^s/_{10}$

- *Elimination of the safety factors for SIF calculations in case of crack depths < $^s/_4$:*

[IAEA 1997]	n_k=2 or $\sqrt{2}$	ΔT= 10 [K]
[SÚJB 1998]	n_k=1	ΔT= 0 [K]
[VERLIFE 2004]	n_k=1	ΔT= 0 [K]

- *Application of the WPS (warm pre-stressing) effect:*

not allowed:	[PNAEG 86] and [SÚJB 1998]	
allowed:	[IAEA 1997]	80% criterion
allowed	[MRK-SChR-2000]	special conditions for 90%
allowed	[VERLIFE 2004]	90% criterion

In all, there are at least ten cases of partly synergistically acting reductions of conservatism in the ETE case:

1. The number of accidents with RPV thermal shock potential is considerable
2. Conservatism of fracture mechanics assessment not proven
3. Thermal hydraulics uncertainties don't allow perfect transient modelling
4. Mixing calculation solvers become very unstable during simulation
5. Remarkable uncertainties with material properties
6. Broad uncertainty of fluence determination
7. Reliance on correct and in time actions by the operator in following EOP's
8. NDT-methods applied in ISI are likely to miss deficiencies
9. Entire cladding must be completely free from unacceptable deficiencies
10. Certain areas of the RPV-wall cannot be inspected.
11. These boundary conditions set deliberately have not been discussed in the detail required for conclusive evaluation.

In summarizing, the following can be stated concerning the Czech approach:

- The applied computer codes for the FEM simulation and the consideration of elastic-plastic material behaviour for the calculation of the temperature and stress fields is considered in accordance with the actual state-of-the art. Once completed all planned analyses using this methodology, the resulting PTS assessment can be considered as a consolidated approach, up to now unprecedented for WWER-1000 reactors.
- The IAEA Guidelines allow the use of postulated crack depths smaller than the normally required ¼ of wall thickness (which is for the WWER-1000 about 50 [mm]) for the case of the NDT-Programme enabling the safe detection of the respective small defect sizes. For this case the IAEA Guidelines require the mandatory use of safety factors: Safety factor 2 for the crack depth or safety factor $\sqrt{2}$ for the stress and ΔT = 10 [K] for the embrittlement induced shift of the critical brittle fracture temperature. In accordance with VERLIFE [PISTORA 2004a] the Czech Experts postulate a crack depth of 20 [mm] only ($^{1}/_{10}$ wall thickness, which is significantly smaller than ¼ wall thickness) but do not apply any the safety factors. (e.g. as required according to the IAEA Guidelines).
- The Czech approach is also deviating from the IAEA Guidelines [IAEA 1997] with respect to results which were not presented about the variations of the crack size and crack geometry:
  - The analyses for very shallow cracks (a < 6 [mm]) and
  - Large cracks (a = 20 [mm] up to ¼ of the wall thickness)[39] and
  - The variation of the aspect ratio to a:c = 1:10.
- The approach taken for integrating the cladding zone into the FE modelling introduces furthermore a reduction of conservatism, not only when excluding elliptical under-clad cracks, but also because assuming a Stress Intensity Factor (SIF) levelling out to SIF = 0 exactly at the cladding/base-material interface does not correspond to reality. This has been reconfirmed by pilot case simulations conducted during the monitoring process.

  The calculation of stress intensity factors taking credit of a defect-free cladding is unusual and raises questions on the conservatism. This approach is based on the assumption of a defect-free cladding with homogeneous characteristics, which as a prerequisite requires perfect quality assurance during manufacture and very sensitive NDT of the cladding. This is contradicted by the fact that significant indications were found in the RPV bottom cladding where bonding failures were observed.
- The FE model represents one half of the reactor pressure vessel. This procedure does not include the stresses from the superposition of the cold plumes, the strain induced curvature of the cylinder and the interaction with the RPV bottom and the RPV head (deformation hindering). This approach is in accordance with the international practice. The simulation using a mesh generated for the complete RPV would represent an outstanding effort.

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[39] According to a remark of M.Brumovsky during the Workshop in Řež the 100% detectability of 10 [mm] would justify the requirement of VERLIFE to calculate only up to crack depths of $^{1}/_{10}$ of the wall thickness, since the VERLIFE requirements concerning the allowability of defects detected during ISI (Appendix XII: “Evaluation of defect allowance in components”) would use a safety factor 2 in the fracture mechanical calculation.

3.6 Load path/fracture toughness curve comparison (incl. WPS effect)

3.6.1 Description of the issue – fundamentals

The demonstration of the RPV integrity is performed in terms of the safety margin between maximum allowable value of the materials critical brittle fracture temperature T_k^a and actual RPV material specific value T_k.

The maximum allowable value of critical brittle fracture temperature is derived from the fracture mechanics calculations of the load paths for each postulated defect in combination with every selected accident transient. These load paths are compared with the fracture toughness curves $K_{Ic}(T-T_k)$; the value of T_k for which the respective $K_{Ic}(T-T_k)$ curve would intersect with the load path (in case of taking credit of the WPS effect: at loads of 80% of the peak stress) is the maximum allowable critical temperature of brittleness for the specific load path (defined accident scenario, defined crack size and position). The minimum value of all these T_k^a values determined for the selected accident scenarios and the postulated crack sizes will define the most critical accident scenario on the one hand and the overall maximum allowable critical temperature on the other hand.

The actual RPV material state is described by the fracture toughness curve where T_k is determined by the formula from the Russian Code using the specified embrittlement coefficient. The experimental data from the surveillance programme are supposed to confirm the conservatism of the embrittlement-coefficient.

The mentioned WPS effect means by definition that crack initiation cannot take place while $dK_I/dt < 0$ (decreasing K_I,). There are worldwide controversial discussions on the applicability of WPS effect and if applied, on the procedure, as will be discussed below.

3.6.2 State-of-the-art requirements and regulations

The application of the WPS effect is a very sensitive issue many experts worldwide argue against – it would take credit of an effect that has only been observed at specimens under laboratory conditions. There is no evidence that components with more complex stress states and more complex thermo-mechanical histories would show this effect, when compared to samples loaded uni-axially. This is one of the reasons why National codes treat the application of the WPS effect in divergent ways.

<u>Russian Federation</u>

At the time of design and manufacture of the Temelín RPV the Russian Code did not allow taking credit of the WPS effect. The new Russian methodology allows taking credit of the WPS effect with a 90% criterion.

<u>United States</u>

The Regulatory Guide 1.154 (US NRC CFR) describes the WPS effect but recommends, not to take credit from WPS for PTS analyses. For licensing purposes, the WPS effect is treated as additional safety margin in the context of a defence in depth strategy (NRC 1992, Shum 1993).

In practice the WPS approach has never been included into PTS analyses in the U.S.:

"It is therefore prudent not to rely on WPS as means of preventing crack initiation. It is a well established phenomenon only for monotonically decreasing K_I, which is not possible to ensure under most accident conditions" [ISKANDER 1986].

"The reason for not including WPS in most of the calculations is that the KI v/s t curves for the shallow flaws are very flat, making it difficult to determine where the maximum is. Furthermore, unforeseen variations in pressure and coolant temperature might exist and defeat WPS." [SELBY 1985a,b]

Germany

Up to 1996, the WPS effect was not included in KTA Rules, following a controversial discussion the WPS effect was implemented into KTA 3201.2, [GERARD 1995, REIMERS 1997].

The German regulations require "basic safety" for materials, manufacture and testing – including a restricted neutron embrittlement susceptibility of the steels, and a limited neutron fluence. Therefore, based on KTA rules structural integrity of the RPV should always be possible to demonstrate without crediting the WPS effect. Germany is the only Western country taking credit of the WPS effect for the structural integrity demonstration of older NPPs with highly embrittled RPV weld materials.[40]

France

The French regulations do not include the WPS effect.

Finland

The deterministic fracture mechanics calculations for the Loviisa-1 RPV have been performed for various sizes of postulated sub-surface and surface cracks in the FE models. The calculation results indicate that the size of the crack tip elements (which were varied between 0,004 and 0,548 [mm]) has a strong effect on the results during the interval of decreased loading with diminishing J-integrals. According to these calculations, which were made for one PTS transient, the effects of warm pre-stressing and shallow crack geometry strongly depend on the transient's pressure history. So far neither the warm pre-stress effect nor the shallow crack effect have been taken into account in the fracture mechanics calculations of the Loviisa RPV and further studies are needed before these effects can be considered [RAJAMÄKI 1993].

IAEA

The IAEA Guidelines [IAEA 1997] state with respect to the consideration of warm prestressing within the PTS analysis: *"In the assessment, warm prestressing could be credited for loads below 0.8 peak stress intensity factor in the continuously decreasing crack loading path, utilizing the assumption, demonstrated by large scale testing, that crack initiation does not occur in the decreasing crack loading path.*

When credit is being given to warm prestress, its applicability in particular for materials with higher embrittlement rate ***should be carefully considered since it may not be fully*** *applicable* ***in the highly embrittled materials****. The national regulatory requirements may not allow to use this approach directly and further justification may be needed."*

[40] The warm pre-stressing approach has been used in the PTS analysis of the NPPs Stade (decommissioning in 2003) and Obrigheim

3.6.3 Current plant status

Workshop presentations by

V. Pištora: Comparison of IAEA, Russian and VERLIFE Methodologies [PISTORA 2004a];

V. Pištora: Summary of Results from Integrity Evaluation [PISTORA 2004b],

V. Pištora, P. Kral: Update of PTS results, Structural analysis [PISTORA 2004d]

The following table shows a summary on the PTS analyses performed by the Czech Experts [PISTORA 2004d]:

PTS group	Scenario	T_k^a [°C]	WPS/ tangent	Weld Nr.	Orientation	$^a/_c$	Crit. time [s]	Crit. point
Pilot study	2SLB	102,8	T	3	axial	0,3	2400	21
	SB32	86,3	W	4	axial	0,3	3000	20
	PSV1	61,2	W	4	circ.	0,7	2100	4
MSLB	SLB1A	126,9	W	3	axial	0,3	1650	21
	SLB1B	108,6	W	3	axial	0,3	1980	21
	SLB1C	111,2	W	3	axial	0,3	2950	21
PRISE	3SGT	66,3	W	3	axial	0,3	3210	20
	SGH1	89,4	W	4	circ	0,7	2330	4
LOCA	C32min	64,6	W	4	circ	0,7	2500	4
	PP210min	76,1	T	4	circ	0,7	4000	4
	H300min	90,1	W	4	circ	0,7	2500	4
	H850	102,7	W	4	circ	0,3	1250	16
PRZSV	PSV43	92,9	W	4	circ	0,7	1730	4
	PSV43B	82,0	T	3	axial	0,3	2300	21
	PSV41	69,9	T	4	circ	0,7	1800	4
	PSV41_rec	64,8	T	4	circ	0,7	1800	4

This table indicates, that

- All accident groups important to be treated in a PTSA were analysed. For WWER-1000 reactors this is the first PTSA with a completeness not achieved up to now.
- The PTS loads for WWER-1000 are extremely high. For a postulated crack depth of only 20 [mm] the resulting T_k^a values are below 70 [°C] in four cases and 3 accident groups, a comparable behaviour is not found with other reactor types, e.g. WWER-440. This is a consequence of the very effective emergency coolant injection systems that are able to compensate large breaks up to ND 850, but induce at the same time a severe thermal shock load at the RPV wall.
- The lowest T_k^a values are found for small to intermediate break sizes where in addition to the thermal shock load a full or partial re-pressurisation of the primary coolant circuit occurs.
- The operator must perform the appropriate emergency operation procedures (EOPs) at the correct moment in order to cope with several accident transients (PSV41) and at the same time avoid brittle failure of the RPV. However, is not international practice to require "guarantied" operational procedures of the personnel, therefore this must be considered a considerable reduction of conservatism in the handling of emergencies.

It is not yet clear, whether the requirement for the T_k^a determination from all calculated $T_k{}^a{}_j$ according to [SÚJB 1998], equation (3.15):

$$T_k^a = min\ \{T_k{}^a{}_j\} - \delta T_k^a$$

$$with\ \ \delta T_k^a = +14\ [K]$$

is included in the VERLIFE methodology.

Applying this requirement the T_k^a value from all calculated PTS transients up to now is 50,6 [°C], which is below the predicted T_k^{EOL} for unit 2.

3.6.4 Evaluation

Literature evaluation concerning WPS:

Crack propagation may be limited by a phenomenon referred to as warm pre-stressing (WPS), which has been demonstrated to some extent on laboratory scale with small specimens and also in a rather large, thick-walled cylinder during a thermal-shock experiment. In such cases, WPS simply refers to the inability of a crack to initiate when KI is decreasing with time while the crack is closing.

While this special situation is encountered during some specific overcooling accidents, caution must be exercised in taking credit of WPS because changes in the pressure can delay or eliminate the conditions for WPS. For instance, a delay in WPS will generally increase the chance of crack-required initiation, and a reversal from dK_I/dt negative to positive can result in crack initiation following WPS.

The report on the thermal shock analysis for the NPP Calvert Cliffs Unit-1using the OCA-P fracture mechanics code states: *"The inclusion of warm prestressing (WPS) in the fracture mechanics analysis would reduce the conditional failure probability several orders of magnitude for many, but not all, of the potential transients. However, because of concerns over the applicability of warm prestressing under certain transient conditions, it was not included in the analysis." [SELBY 1985a]. In addition, in the frame of a thermal shock analysis concerning NPP H.B. Robinson Unit-2 WPS was not applied because "unforeseen variations in pressure and coolant temperature might exist and defeat WPS."* [SELBY 1985b].

Experimental studies on the WPS effect have been enhanced in the last years because the demonstration of RPVI for older PWRs with high neutron embrittlement (Germany: NPP Obrigheim and NPP Stade[41]) is not possible without taking credit of the WPS effect.

Several experimental investigations on the WPS effect on ferritic pressure vessel steels 10 MnMoNi5-5 and 17MoV8-4 are reported. Increased fracture toughness at –120 [°C] and 20 [°C] was observed, dependent on the level of the pre-loading ([ROOS 1997], [BLUMENAUER 1997]).

With respect to transferability of the results from the small samples to the component, some authors [BLUMENAUER 1997] indicate that in the component different constraint conditions and other residual stresses have to be considered. The samples with simulated embrittlement showed a lower increase of fracture toughness induced by WPS.

[41] decommissioned in 2003

Recent experimental results on WWER steels on WPS reported from Ukraine [YASNIY 2003] show that the static blunting was observed but the preloading necessary for this effect would be two to three times that of a hydro test, which is not possible[42].

Due to the fact that neutron embrittlement is causing a different microstructure state compared to heat treatment induced embrittlement it could also be expected that the positive effect of WPS might be significantly reduced.

Further research efforts will be concentrated in the R&D program SMILE ("Structural Margin Improvements in Aged-Embrittled RPV with load History Effect"), which is part of the Fifth Framework Programme of EURATOM.

Summarizing, it is obvious that the applicability of the WPS effect is still controversial due to theoretical and experimental uncertainties, especially with respect to the transferability of laboratory results for temperature/load transients that might significantly differ from the real situation in the component and the temperature/pressure history during a realistic PTS event. It is also obvious that the application of the WPS effect is reducing the safety margin significantly, which is in contradiction with the uncertainties concerning the embrittlement prediction for radiation-sensitive RPV steels.

<u>Discussion of the Czech PTS analyses</u>

With respect to the concept, the methodology and the applied computer codes, the presented Czech approach for PTS analyses (presented as part of VERLIFE), appears to be in accordance with the state of the art.

All important accident groups were analysed in this WWER-1000 PTSA.

The PTS loads for WWER-1000 RPVs are extremely high. For a postulated crack depth of only 20 [mm], the resulting T_k^a values are below 70 [°C] in four cases and 3 accident groups. This is caused by the very effective emergency cooling injections systems, which compensate large breaks up to ND 850 but induce at the same time a severe thermal shock load at the RPV wall.

The lowest T_k^a values are found for small to medium break sizes where in addition to the thermal shock load a complete or partial re-pressurisation of the primary coolant circuit occurs.

For several accident transients, operational procedures (EOPs) have to be entered at a precise point in time for the operator to be able to master the accident and to avoid brittle failure of the RPV.

The evaluation of the results of thermal hydraulic and mixing calculations, the temperature and stress fields and calculation of the stress intensity factors indicates that there is a need of thorough discussion:

Even neglecting the safety factors (see below) the resulting T_k^a values are extremely low which indicates the tendency that the reactor pressure vessels of WWER-1000 reactors experience much higher thermal shock loads than the reactor pressure vessels of WWER-440 reactors. The modification of the WWER-440/type 230,where only intermediate breaks could be compensated, for the modern version WWER-440/type 213 that was supposed to allow compensation of a large break (ND 500: DEGB of the primary coolant loop) did already result in a considerable increase of thermal shock loads. The increased capability of the emer-

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[42] "Warm pre-stress method deals with the loading and unloading of the structure cracked element at the temperature that exceeds the operating temperature. The raising of critical SIF under static blunting loading after overloading of 15Ch2MFA steel specimens after heat treatment, which simulates the radiation embrittlement of the steel by the end of the operating life, can be in 3 times higher than compared with the static fracture toughness of the material. But the problem as to the application of the method for RPV is that in order to obtain sufficient raising of brittle strength, the structure should be subject to the inside stress which is in 2 – 3 times higher than of hydraulic test. It is usually impossible".

gency cooling systems necessary for compensation of large breaks in the primary coolant system causes an increase of the amount of cold coolant injected into the downcomer during emergency operation. This is induces three effects:

1. The reactor pressure vessel wall is cooled down very rapidly to the temperature range 20 ÷ 80 [°C], which might already be the range of ductile-brittle transition of the steel taking into account the neutron induced embrittlement of the steel. The temperature dependence of the fracture toughness shows this effect quantitatively.
2. The high capability of the emergency cooling systems causes extreme temperature transients inducing steep temperature gradients over the pressure vessel wall thickness. Due to the incomplete and delayed mixing of the injected cold water with the high temperature reactor coolant circumferential and axial temperature gradients develop (cold plumes).
3. For small and intermediate breaks, the rapid compensation by the high capability of the emergency cooling systems causes an early and rapid re-pressurisation of the primary circuit, which adds to the thermal shock load.

For the WWER-440/type 230 reactor the T_k^a values were in the range of 140 ÷ 200 [°C], for the WWER-440/type 213 reactors in the range 90 ÷ 160 [°C]. It is obvious that for the WWER-1000 reactors the T_k^a values are – even with the incorrect application of safety factors – in a range with a lower bound significantly below 90 [°C].

In the following, the quantitative effects of the elimination of the safety factors required by the IAEA Guidelines [IAEA 1997] (which has to be considered as a change of the safety philosophy) shall be discussed in detail for the selected accident C2 (simultaneous break of two steam generator heating tubes):

The PTS analysis results from this accident transient are discussed in the Workshop presentation by V. Pištora: "Summary of Results from Integrity Evaluation" [PISTORA 2004b].

On page 28 the calculated load path (stress intensity factor versus temperature at the crack tip) is shown, the SIF (stress intensity factor) maximum is reached at 92 [MPa.m$^{1/2}$]. The application of the WPS criterion with 90% of the maximum (deviating from the IAEA Guidelines) the resulting T_k^a value is according to [PISTORA 2004b] 66,3 [°C]. The use of the 80% level, as required by IAEA would result in a T_k^a value of 50 [°C] at a SIF value of 73,6 [MPa.m$^{1/2}$].

The IAEA Guidelines [IAEA 1997] allow the use of postulated crack depths smaller than the normally required ¼ of wall thickness (which is for the WWER-1000 about 50 [mm]) for the case of the NDT-Programme enabling the safe detection of the respective small defect sizes. For this case, the IAEA Guidelines require the mandatory use of safety factors:

- Safety factor 2 for the crack depth or safety factor √2 for the stress, **and**
- ΔT = 10 [K] for the embrittlement induced shift of the critical temperature of brittleness.

In accordance with VERLIFE provisions, the Czech Experts postulate a crack depth of 20 [mm] only ($^1/_{10}$ wall thickness, which is significantly smaller than ¼ wall thickness) and do not apply the safety factors recommended by the IAEA Guidelines.

The application of the safety factor √2 as required by the IAEA Guidelines for the accident transient C2 would result in a maximum allowable critical brittleness temperature T_k^a of 43 [°C] in case of the 90% WPS-criterion and 25 [°C] in case of applied 80% WPS-criterion. The safety factor ΔT = 10 [K] is required in addition – it is supposed to cover the uncertainties of the embrittlement prediction – it would result in values for the maximum allowable critical temperature of brittleness T_k^a of 33 [°C] or 15 [°C], respectively.

These values of the maximum allowable critical temperature of brittleness T_k^a are significantly below the predicted EOL values for the critical temperature of brittleness T_k^{EOL} of 38 [°C] for unit 1 and 51 [°C] for unit 2 of NPP Temelín, as determined using the specified embrittlement coefficients of the Russian Code. Considering the fact that 50% of the embrittlement is already reached within the first five years of operation it is clear that the critical conditions will occur significantly before the projected EOL.[43]

In addition it has to be considered that the neutron induced shift of the critical temperature of brittleness as predicted by the specified embrittlement coefficients of A_F = 23 for the base metal, and A_F = 20 for the weld metal might not be conservative (see also chapter 4.2).

These facts might explain the interest of countries operating WWER-1000 reactors to reduce the safety standards as commonly elaborated within the IAEA Guidelines [IAEA 1997] by a significant reduction of the safety factors.

A detailed PTSA – given an appropriate sorting out of results – must allow amongst other points to determine the required sensitivity of the defectoscopy applied (generally ultrasonic testing).

This topic is even more urgent, in case the PTSA is not conducted according to the historical standard procedures by evaluating a crack with ¼ of the wall thickness and where only small cracks are demonstrated to be stable. In such a case, the PTSA must elaborate clearly on the entire spectrum of cracks with the instability criterion fulfilled at the crack-front. Moreover, this topic again becomes even more urgent, in case safety factors stipulated by IAEA-Guidelines are suppressed.

Regrettably enough, the PTSA for Temelín was not elaborated to such detail in this sense – in order to be able to give an answer to the question for the smallest critical crack and that way for the sensitivity required from defectoscopy. At the same time the results were not communicated during the presentations at the Workshop in such manner, that one could derive quantifications from it.

Frequently – and evidently also within the present PTSA for Temelín NPP the analyses are limited to determine the earliest point in time where the criterion for instability is fulfilled, a finding that is suitable to derive T_k^a. Once this value is determined, the subsequent investigations are ceased. Computing results from other PWRs show, however, that as a rule short time after instability is reached for a large crack (e.g. 200 [mm] deep crack), also smaller cracks become instable during the subsequent accident progression in time. Their T_k^a-values are lower indeed, than those of the main crack, but during the preset accident progression, they become instable as the main crack does – in general somewhat earlier. Consequently, a whole range of crack depths and crack axes ratios characterises every accident sequence, for which individually determined instability conditions could be fulfilled. With increasing RPV operation time (neutron embrittlement), this range grows in particular into the range of smaller crack depths; this must be characterised. Deriving requirements for the sensitivity of defectoscopy from PTSA is impossible or at least is of limited significance when this procedure is neglected.

[43] The situation is even more critical for other WWER-1000 reactors that do not have as low T_{k0} values as ETE.

3.7 Conclusions concerning PTS analysis

The presented Czech approach for PTS analyses (part of the VERLIFE) with respect to the concept, the methodology and the applied computer codes are considered to be in accordance with the state of the art.

Thermal hydraulic analyses:

- The RPVI/PTS TH-analyses can be considered comprehensive in the sense of the [IAEA 1997] recommendations and should have provided a sound basis for the selection of candidate transients for the mixing and heat transfer calculations to be conducted subsequently. The [IAEA 1997] Appendix IV contains a special list of initiating events recommended for consideration with WWER-1000 NPPs and the number of events analysed as well as the types of transients treated for ETE suggest, that a "full" PTS related thermal hydraulic analyses programme was performed. It is worth mentioning, that the withdrawals implied in the calculations by using assumptions, which cannot be understood as conservative; attention should be devoted to this.
- Evidently all thermal hydraulic calculations work has been performed with state of technology computer codes, and at least some benchmark calculation were made with a different code in order to establish confidence into the methods selected.
- Results, namely those representing the most severe transients, are by all means comparable to those considered representative for WWER-1000 installations according to current knowledge. The uncertainties involved in simulating events exhibiting strong gradients during their course have not been presented. These omitted could contribute to drawing too favourable a picture.
- In some instances there were doubts as to whether the time frame the simulation observed might have caught the essence of the loading to the RPV, since re-pressurization might just not have taken place that early.

Mixing calculations and heat transfer issues:

- The mixing calculations fort the accident transients within the PTS analyses performed appear to be in accordance with the state-of-the-art in international practice and comparable to calculations for other WWER reactors.
- The mixing simulation calculations are based on the application of accepted, modified computer codes REMIX/NEWMIX and CATHARE for the simulation of PTS mixing transients. The "TH-boundary conditions" are taken from results obtained with simulation codes extensively used and benchmarked in the WWER-1000 simulation environment. This conforms to current practice in Europe.
- The mixing simulations are performed in the respective domains of both the REMIX/NEWMIX and CATHARE code. They appear to cover a broad, but not the entire spectrum of transients involving single-phase and two-phase coolant flow mixing situations in the downcomer. Restrictions in the codes do not allow mixing simulation of multiple "plumes" of additional cold coolant with reactor coolant in the downcomer in the REMIX/NEWMIX case, and because of the 2D-representation of CATHARE do not allow simulation of radial stratification in this case. For all these additional simulation requirements reasoning for omission should be provided, otherwise the need for additional reconfirmation of conservatism is evident.
- The heat transfer properties are derived from RELAP or ATHLET results, as it was explained. The assumptions have not been explained to the extent, which heat transfer mechanisms have to be assumed for the different flow regimes, what boundary layer influence can be implied and what heat transport phenomena can really be respected by RELAP or ATHLET. Again, the transfer of heat properties distributed over the model according to the discretisation could not be verified. The Czech Experts provided additional

information intended to endorse the argument of abundant conservatism exceeding comparable PTSA application cases. The review of these results will have to be performed using a set of pilot calculations suitable for comparison. The identified lack of conclusive coverage of this aspect at the Workshop should be overcome.

<u>FEM calculations and Fracture Mechanics evaluation:</u>

- The applied computer codes for the FEM simulation and the consideration of elastic-plastic material behaviour for the temperature and stress field calculations is considered to be in accordance with the actual state-of-the art. Once completed all planned analyses using this methodology, the resulting PTS assessment can be considered as a consolidated approach, up to now unprecedented for WWER-1000 reactors.
- The methodology of the SIF calculations where the cladding is included into the FE model, the approach taken by the Czech Experts (calculation of only semi-elliptical cracks) is less conservative than the internationally accepted practice of postulating semi-elliptical surface cracks without taking credit of the cladding for the stress intensity factor calculation. This approach needs additional monitoring and has to be based on a qualified and perfect NDT for the cladding.
- The IAEA Guidelines allow the use of postulated crack depths smaller than the normally required ¼ of wall thickness (which is for the WWER-1000 about 50 [mm]) for the case of the NDT-Programme enabling the safe detection of the respective small defect sizes. For this case the IAEA Guidelines require the mandatory use of safety factors: Safety factor 2 for the crack depth or safety factor $\sqrt{2}$ for the stress and $\Delta T = 10$ [K] for the embrittlement induced shift of the critical brittle fracture temperature. In accordance with VERLIFE [PISTORA 2004a], the Czech Experts postulate a crack depth of 20 [mm] only ($^1/_{10}$ wall thickness, which is significantly smaller than ¼ wall thickness) but do not apply any safety factors (e.g. as required according to the IAEA Guidelines).
- The Czech approach is also deviating from the IAEA Guidelines [IAEA 1997] with respect to results not presented about the variations of the crack size and crack geometry:
  - The analyses for very shallow cracks (a < 6 [mm]) and
  - Large cracks (a = 20 [mm] up to ¼ of the wall thickness) and
  - The variation of the aspect ratio to a:c = 1:10.
- The approach taken for integrating the cladding zone into the FE modelling introduces furthermore a reduction of conservatism, not only when excluding elliptical under-clad cracks, but also because assuming a Stress Intensity Factor (SIF) levelling out to SIF=0 exactly at the cladding/base-material interface does not correspond to reality. This has been reconfirmed by pilot case simulations conducted during the monitoring process.
- The FEM modelling represents only one half of the reactor pressure vessel. This procedure does not include the stresses from the superposition of the cold plumes, the curvature of the cylinder and the interaction with the RPV bottom and the RPV head (deformation hindering). Nevertheless, this approach is in accordance with the international practice. The mesh generation for the complete RPV would represent an outstanding effort.

With respect to the PTSA:

- All accident groups important to be treated in a PTSA were analysed. For WWER-1000 reactors this is the first PTSA with a completeness not achieved up to now.
- The PTS loads for WWER-1000 are extremely high. For a postulated crack depth of only 20 [mm] the resulting T_k^a values are below 70 [°C] in four cases and 3 accident groups, a comparable behaviour is not found with other reactor types, e.g. WWER-440. This is a consequence of the very effective emergency coolant injection systems that are able to compensate large breaks up to ND 850 but induce at the same time a severe thermal shock load at the RPV wall.

- The lowest T_k^a values are found for small to intermediate break sizes where in addition to the thermal shock load a full or partial re-pressurisation of the primary coolant circuit occurs.
- The operator must perform the appropriate emergency operation procedures (EOPs) at the correct moment in order to cope with several accident transients (PSV41) and at the same time avoid brittle failure of the RPV. However, is not international practice to require "guarantied" operational procedures of the personnel, therefore this must be considered a considerable reduction of conservatism in the handling of emergencies.
- Some accidents (PSV43) have not been calculated until to the point of applicability of the 90% criterion.
- In some cases the definition of the accident transients cannot be considered the most critical one: In accident group PRZ SV the total loss of offsite power has not been included although required by the IAEA Guidelines. Including total loss of offsite power would induce a re-pressurisation in the primary circuit following the re-closure of the pressuriser safety valve.

With respect to the safety factors required by the IAEA Guidelines, it has to be stated:

- The VERLIFE methodology as applied by the Czech Experts for NPP Temelín uses only postulated crack depths of 20 [mm] ($^1/_{10}$ wall thickness, which is significantly smaller than ¼ wall thickness) and applies no the safety factors at all.
- The VERLIFE methodology as applied by the Czech Experts for NPP Temelín is applying the 90% WPS criterion although the IAEA Guidelines recommend the 80% level, if applied at all. This modification further reduces significantly conservatism, which contradicts the need to compensate for uncertainties in embrittlement prediction for radiation-sensitive RPV steels.
- The applicability of the WPS effect is still judged controversial in the international community due to theoretical and experimental uncertainties, especially with respect to the transferability of laboratory results for temperature/load transients that might significantly differ from the real situation in the component and the temperature/pressure history during a realistic PTS event.
- For the accident transient C2 as an example: The PTS analysis including the application of the safety factor $\sqrt{2}$ and the additionally required safety factor $\Delta T = 10$ [K] as required by the IAEA Guidelines and the application of the 80% WPS-criterion as recommended by the IAEA Guidelines would yield a maximum allowable critical temperature of brittleness T_k^a of 15 [°C]. This value is by far lower than the one predicted EOL for the critical temperature of brittleness T_k^{EOL}.
- The requirement in [SÚJB 1998] to apply a safety margin δT_k^a for the final assessment of T_k^a has not been included in the Czech presentation. It is not clear whether this safety margin has also been eliminated in the VERLIFE methodology.

3.8 Issues of further interest, monitoring items concerning PTS analysis

- Since the present RPVI work did not explicitly take into consideration cold overpressurization and outage situations, these two topics remain as recommended issues for a further monitoring.
- Mixing simulation development and all related additional PTSA relevant activities are of special interest. With regard to heat, transfer assumptions verification a need for clarification has been identified.
- The development of 3-D applications is recommended be followed and as soon as they will have reached acceptance for licensing the respective mixing simulation results a reconsideration is recommended be stipulated, because of the key function mixing and the related heat transfer behaviour plays in the PTS events.

The PTSA for Temelín was not elaborated to such detail to enable the Austrian Experts' Team to give an answer to the question for the smallest critical crack size and thus the required sensitivity for the defectoscopy, giving reason for the following monitoring items:

- The analysis should be realised using a broader range of crack depths, crack shapes and crack orientations. Up to now the number of variants evaluated within the model was limited to 16 each: 4 for crack depth, 2 for the crack axes ratio, and two for the crack orientation. The crack depth was restricted to 8 [mm] for the lower and 20 [mm] for the upper boundaries and the crack axes ratio was limited to 2 values, namely 0,3 and 0,7.
- The analysis should become more refined. In the present PTSA the 4 mm step on the crack depth appears to be too coarse.
- The effects of the cladding included into the SIF calculations should be reviewed for acceptability together with the FE methodology applied.
- A complete stability analysis until the end of the transient should be conducted for all crack variants.

4 SURVEILLANCE PROGRAMME – MATERIAL EMBRITTLEMENT

Areas of Monitoring

No	VLI/VLI group description
2	**DESIGN & MANUFACTURING**
2.4	**MATERIAL SELECTION, PROPERTIES AND EMBRITTLEMENT MANAGEMENT**
2.4.1	**Material selection: RPV design requirements, RPV steel manufacturing, mechanical properties, RPV steel “design-modifications”**
1	Was it a licensing precondition to fulfil all the RPV design requirements by the main constructor? How many deviations are documented, which were the most remarkable and by which procedures were they accepted?
2	Do the RPV-material mechanical properties fulfil the specifications of the main constructor design in accordance to the Russian Code requirements?
3	Which requirements specified the steel manufacturing? How did the manufacturing processes performed take place according to the Russian Code?
4	Has there been any RPV design modification during manufacture? Were there any additional requirements to be fulfilled in order to verify functionality of the RPV while preparing and/or following the modification(s)?
2.4.2	**Embrittlement: Embrittlement prediction methods, application and results**
1	Which embrittlement prediction methods were used and are there predictions in accordance with the Russian Code? Were there modifications (deviations from the Russian code) introduced into the National Code requirements?
2	Which provisions are foreseen in case the first surveillance results exceed the specifications of the Russian Code?
2.4.3	**Embrittlement: Surveillance programs forWWER-1000**
1	Have systematic evaluations been done regarding the performance of surveillance programmes for WWER-1000?
2	Have any weaknesses been recognised regarding the surveillance programmes for WWER-1000, which have not been addressed at Temelín? Have any specific needs for extending the range of surveillance provisions been identified and changes implemented?
2.4.4	**Embrittlement: Surveillance monitoring provisions for ETE RPV material properties**
1	Is there a documented plant programme for preventive maintenance and surveillance? Is the implementation of this programme properly supported by plant procedures?
2	Which provisions have been made for the monitoring of the ETE RPV material properties’ initial state and the properties’ development with the operation history as well?
3	Is this monitoring program identical for both units? If not what are significant differences?
2.4.5	**Embrittlement: Comparison with Russian Code regulations, IAEA Guidelines**
1	Are the modifications of the implemented surveillance programme compatible with the Russian Code requirements? What about the fluence correction and prediction comparison?
2	Is the number of specimen for samples of base metal, weld metal, HAZ and the kind of specimen – tensile, Charpy V-Notch, fracture mechanics – supplied to the surveillance programme at ETE in accordance with Russian Code requirements?
3	Are all the surveillance programme features in accordance with IAEA Guidelines?
4	Is the number of specimen for samples of base metal, weld metal, HAZ and the kind of specimen – tensile, Charpy V-Notch, fracture mechanics – supplied to the surveillance programme at ETE in accordance with the IAEA Guideline recommendations?

No	VLI/VLI group description
2	**DESIGN & MANUFACTURING**
2.4.6	**Embrittlement: Comparison with EU widely accepted requirements**
1	Have any comparative evaluations been performed with respect to the surveillance programmes of EU state-of-the art?
2	Is the number of specimen for samples of base metal, weld metal, HAZ and the kind of specimen – tensile, Charpy V-Notch, fracture mechanics – supplied to the surveillance program comparable to European practice?

Areas of Monitoring

No	VLI/VLI group description
3	**OPERATION & MAINTENANCE**
3.4	**MATERIAL SELECTION, PROPERTIES AND EMBRITTLEMENT HISTORY**
3.4.1	**Material selection, properties and verification: RPV steel, mechanical properties, RPV steel design modifications, requirements and RPV as manufactured**
1	Have the RPV materials surveillance samples properties been verified to be representative for ETE-1 and ETE-2?
2	Are the mechanical properties of the unirradiated surveillance samples comparable with the measured data from test coupons and qualification test coupons?
3	Are the material properties of RPV materials similar for ETE-1 and ETE-2?
4	How are uncertainties involved in irradiation minimization handled in analysis and in the surveillance program and PTSA?
3.4.2	**Surveillance program in ETE-1 and ETE-2**
1	What is the withdrawal schedule for surveillance samples in ETE-1 and ETE-2?
2	What kinds of samples are included in the first set of surveillance specimen to be withdrawn from their irradiation capsules (Charpy, tensile, fracture mechanical)?
3	When are the first results on neutron embrittlement data expected?
3.4.3	**Monitoring of ETE RPV material properties and embrittlement: Surveillance Program and results**
1	Has the first capsule withdrawal already been performed for ETE-1? Are there already results concerning embrittlement?
2	How are corrections of the reactor history been treated at Temelín in the context of determining neutron fluence? How are axial flux density variations due to fuel burn-up history and control rod movements reflected in determining neutron fluence? How are these factors used in evaluating the surveillance specimens removed from the reactor?
3.5	**MATERIAL EMBRITTLEMENT HISTORY VERIFICATION AND CONSEQUENCES**
3.5.1	**ETE – PTSA implementation, verification**
1	Was any additional instrumentation dedicated to PTS/RPVI installed? What is the functionality of the relevant instrumentation in station blackout conditions? Was any operational aid developed to provide for unambiguous online status information? Are there provisions for early alerting the MCR personnel?
2	Is all information needed for PTS/RPVI management available in the MCR and at the on-site emergency centre (TSC) as well? Is the information provided for PTS/RPVI management user friendly?
3	Are there provisions in place for minimizing human errors during PTS events?

No	VLI/VLI group description
3	**OPERATION & MAINTENANCE**
4	What was the systematic approach to identify and overcome limitations (of power supply, coolant media, etc.) associated with equipment operated according to the PTSA under accident conditions?
5	Whenever challenges by PTS evolve to RPVI, are there any threats identified that can affect the MCR personnel?
6	What quality assurance programme is implemented for reactor surveillance dosimetry at Temelín? In this context, how does the programme treat the calibration of the surveillance system against national standards? What other types of quality assurance checks are performed (e.g., routine surveillance of neutron dosimetry information, inter-laboratory comparisons) and how often? How is feedback from these quality assurance checks into the reactor surveillance dosimetry programme accomplished?
7	By what means is the accuracy and reliability of neutron dosimetry for the RPV maintained? To what extent does the plant combine the surveillance capsule and cavity dosimetry measurements with plant-specific neutron transport calculations?
8	Which damage correlation studies and experimental work for trend curves development are or will be performed for the Temelín RPVs?
9	By which means does the plant accomplish an integration of knowledge of material property changes with neutron dosimetry results?

4.1 Surveillance Programme Bases

4.1.1 Description of the issue – fundamentals

In principle, the material properties degradation due to neutron irradiation can be predicted based on the knowledge of experimental test results on RPV steels performed over the years during development of the specific type of reactors. The results of this broad experimental background have been used for the definition of material requirements in the National Codes. Besides this, in practically all countries the material degradation (embrittlement) of the plant specific RPV materials (except for the very first NPPs) is monitored throughout the operational lifetime by the so-called surveillance programmes.

The behaviour of the specific steel of an individual reactor pressure vessel under irradiation conditions during operation has been monitored since the early 70^{ies}; it turned out that in many cases the radiation embrittlement was significant although only minor changes of the ductile-brittle transition temperature and the of upper-shelf toughness had been expected. The observed strong embrittlement was attributed to relatively high copper and phosphorus impurity levels in the respective RPV steel or weld material.

Since that time surveillance programs with representative samples of the vessel material are mandatory (representative samples are made using oversize cuts of the ring base material and special welds, manufactured using identical base material and weld electrodes, and identical manufacturing conditions as for the RPV). The irradiation capsules for the surveillance samples have to be located in the RPV so that the neutron flux at the sample location is higher than at the vessel wall in the belt region in order to reach an accelerated irradiation that allows prognostic information on the embrittlement behaviour. The so-called “lead factor” is the relation of the flux at the surveillance sample position and the flux at the vessel wall in the region of the active core.

The capsules are withdrawn at regular time periods for experimental destructive testing. This procedure provides an accelerated irradiation condition for each specimen set that allows predictive data on the embrittlement. The discovery of a possible fluence rate effect (high embrittlement for lower neutron fluxes compared to higher flux for identical neutron fluence values) caused the need to restrict the lead factor to about 2 in order to avoid falsification of the results.

4.1.2 State of the art requirements and regulations

Most Western National Code requirements concerning the surveillance programmes are based on the U.S. requirements [U.S. Code of Federal Regulation, Title 10, part 50, Appendix H]. As in other chapters, the regulations in the United Kingdom are referred to demonstrate the philosophy behind the rather liberal looking way of licensing nuclear power plants.

Russian Federation

According to the regulatory requirements for the materials to be included in the surveillance program for the WWER-1000 the base material of the shell with maximum (P+0,2Cu) [%wt] has to be used. The weld material to be used is defined as the weld with maximum irradiation.

The type of specimens required for WWER-1000 surveillance programmes include Charpy, tensile, fracture mechanics and fatigue samples.

The minimum number of specimens and the orientation is regulated.

United States

The Code of Federal Regulations includes an Appendix on “Reactor vessel material surveillance program requirements” prescribing the materials to be included (base metal, weld metal, HAZ from the beltline region), the number and type of specimens (Charpy, tensile) and the location of the capsules near the inside vessel wall, so that the irradiation conditions duplicate the conditions experienced by the beltline region of the RPV.

Germany

The German Code defines the surveillance programme in KTA 3203 “Monitoring the radiation embrittlement of materials of the reactor pressure vessel of light water reactors”. The requirements are defined for the materials to be included, the sampling locations, the orientation of the samples, number and type of specimens (Charpy, tensile, fracture mechanic) the location of the surveillance capsules inside the vessel including limits for the lead factor, and withdrawal schedule.

France

The regulatory requirements from 1997 define a surveillance programme based on the U.S. regulations; they were codified in the RSEM Code. The surveillance practice varies for the different types of NPP units with respect to the definition of the beltline region, the minimum number of specimens and the type of fracture mechanic samples.

United Kingdom

In accordance with the non-prescriptive nature of the U.K. regulations no specific requirements are established, but the “Safety Principle P101” specifies:

“Provisions should be made of periodic measurements of relevant properties of fully representative materials and parameters relevant to the design of the plant where such properties or parameters could change with time and effect safety.” [NICHOLSON 1994].

Surveillance programmes in WWER-1000/320 plants – surveillance programme in the NPP Temelín

Workshop presentation: J. Žďárek, M. Brumovský : Surveillance programme status

The surveillance programmes in WWER-1000/320 plants had severe deficiencies with respect to the reliability of the resulting data.

The IAEA stated within an extrabudgetary programme concerning WWER-1000/320 plants [IAEA 1996]:

"The exposure of the WWER-1000 vessel wall to fast neutron flux is in the same range as in the western PWR vessels of the same vintage. The calculated maximum end-of-life fluence (after 40 years) is 5,7 x 10^{23} [1/m^{-2}], E > 0,5 [MeV]. However, at most of the plants, the irradiation embrittlement could progress faster than anticipated due to a higher Ni concentration of up to 1,9 wt.% in vessel beltline area welds.

Surveillance specimen, representing base weld and heat affected zone metal of one core shell, should provide for monitoring of mechanical properties and brittle fracture temperature changes due to irradiation and thermal ageing. The containers with specimens to monitor irradiation embrittlement are, in a standard design, placed on the top of the thermal shield top shell, where the temperature and neutron field are considerably different from the vessel wall. Thus, the data from the surveillance specimen to monitor RPV wall irradiation embrittlement are not fully applicable to vessel wall and can support the vessel status prediction to a limited extent only, if at all. There are only limited relevant data from R&D programmes available, mostly based on irradiation at 270 [°C] in WWER-440/213 empty surveillance positions or for steels with Ni contents less than 1,5%wt."

These severe deficiencies are described in more detail in [AMAEV 1999, BRUMOVSKY 2001]:

- Large gradient of the neutron flux at the specimens of one capsule due to improper arrangement in an reactor[44]
- Uncertainties of neutron fluence determination due to the lack of neutron monitors
- Low lead factor due to the location of the irradiation containers[45]
- Possible γ-heating of the surveillance specimens during irradiation
- The temperature at the capsule location is higher than the temperature at the RPV wall in the core region (about 15 ÷ 20 [K] higher)[46]
- The exact irradiation temperature is unknown due to the lack of reliable temperature monitors

[44] Thus it is practically impossible to find six containers with similar fluence, 12 specimens are required for one Charpy curve to determine a single value of the brittle fracture temperature for this specific neutron fluence.

[45] For the upper floor of the containers the lead factor was even lower than one.

[46] Higher irradiation temperature caused lower embrittlement due to annealing effects; thus the results cannot be applied for RPV embrittlement prediction.

4.1.3 Current plant status

Surveillance programme in the NPP Temelín

Workshop presentations: M. Brumovský, J. Žďárek: Reactor Pressure Vessel Integrity (RPVI) Assurance Approach, J. Žďárek: Surveillance Programme Status,

According to [BRUMOVSKY 2001] the surveillance program for NPP Temelín has been modified so that this surveillance program can be considered the first reliable surveillance program for WWER-1000/V320 reactors. The characteristic changes of the Temelín surveillance program are the following (information from [BRUMOVSKY 2001] and [ZDAREK 2004b]):

- Flat containers (200 x 200 x 25 mm inner dimensions) containing all specimens of one irradiation set.
- Containers are located in special holders welded on the inner surface of the RPV approx. 400 mm below the centre line of the belt region (in [BRUMOVSKY 2001] the information was: about 900 mm below the active core centre) with a lead factor of 2 to 3 (in [BRUMOVSKY 2001] the information was: lead factor 1,5 to 2).
- In the presentation [ZDAREK 2004a] the withdrawal scheme is after 2, 6, 10, 18, 28 + x years: one container for thermal annealing effect, and one container for re-embrittlement rate effect.
- In [BRUMOVSKY 2001] the planned withdrawal schedule was 2, 6, 10 and 20 years, + two sets for potential annealing purposes (in the same presentation there is also the information, that the planned removal is after 2, 4, 12, 18 and 24 years).
- 2 containers are located in the upper part of the vessel in the outlet water region with the purpose to study thermal ageing (planned removal 14, 34 years; in [BRUMOVSKY 2001]: 8, 24 years).
- The materials foreseen for irradiation within the surveillance program:
  - Base materials from the central beltline ring,
  - Weld metal from the beltline weld (weld Nr. 3)
  - Heat affected zone material
- Neutron fluence monitors and irradiation temperature monitors are supplied and will allow a reliable determination of the irradiation temperature and the irradiation fluence.

The specimens in the containers to be irradiated will be tested by the following methods: mini-tensile tests, Charpy impact test, pre-cracked Charpy for dynamic fracture toughness determination, fracture toughness.

The first container was withdrawn on Sunday, May 9th, 2004; the transport to NRI will occur in June 2004. The results of the analysis of the irradiated samples will be available next year.

4.1.4 Evaluation

The modified surveillance program has eliminated the described deficiencies of the original WWER-1000 surveillance programmes, esp. with respect to irradiation temperature, neutron flux and fluence at the sample location. The embrittlement information of the Temelín irradiated samples will provide the first reliable data on WWER RPV material embrittlement.

The disadvantage of the modified surveillance programme results from the weldments between the irradiation capsule holders and the vessel wall 400 mm below the centreline of the beltline region and between the upper nozzles. These locations will not be accessible for NDE measurements.

The discussion during the workshop has shown that the Czech Experts had to decide between the alternatives

- Reliable surveillance data but restricted NDT accessibility, and
- NDT accessibility but deficiencies in the surveillance results.

The Czech Experts' decision was in favour of the reliable embrittlement information.

In spite of the fact that the surveillance data will have the required reliability it has to be stated, that each removal of capsules will contribute with a single value in the embrittlement versus fluence (operation time) plot. It is clear that the surveillance programme of one power plant cannot provide the statistical basis required to reliably predict the embrittlement progress throughout service life. In case of observed stronger embrittlement data from the surveillance samples compared to the specified predictions the results would be further indications of the non-conservatism of the specified values. Nevertheless, for reliable predictions a broad statistical basis of reliable data[47] is necessary.

4.2 Material properties

This chapter provides an overview on the basic material properties and neutron embrittlement of the RPV materials.

4.2.1 RPV steel and properties

4.2.1.1 Description of the issue – fundamentals

Reactor pressure vessels are made of ferritic-bainitic steel, at the inside surface cladded with stainless steel for corrosion protection. Only older WWER-440/230 pressure vessels were manufactured without cladding at the inside.

The Russian chromium-molybdenum-vanadium mild steel type 15Kh2NMFA-A is the WWER-1000/320 RPV base metal, whereas the weld material is of the Sv-12Ch2N2MAA type.

In order to judge the applicability of specific steels for the manufacture of pressure vessels specific material data characterising the mechanical properties are defined, such as Young modulus, yield strength, ultimate tensile strength, ductile-brittle transition temperature and upper-shelf toughness. Using these characteristics it is possible to demonstrate based on theoretical elastic calculations that a certain component has the necessary strength for the postulated loading conditions.

Tensile properties, such as Young modulus, yield strength and ultimate strength are determined by destructive methods using tensile testing machines. The ductile-brittle transition temperature is usually determined by Charpy test methods, which use the measurement of the absorbed energy of a V-notched sample at break through impact at a certain temperature.

Reference transition temperature RT_{NDT} or T_K, respectively

Using Charpy test measurements it was necessary to define a measure for the transition from brittle state at low temperatures to the tough state at high temperature. This reference temperature is not a physical property but an agreed definition of a useful measure for the related physical property "ductile-brittle transition temperature (DBTT)"; DBTT is a characteristic property for the state of the material after any physical treatment.

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[47] It has to be kept in mind that the Temelin surveillance program is the first WWER-1000 surveillance program that will deliver reliable surveillance data.

The definition of this reference temperature is based on the experimentally derived Charpy curve $C_v(T)$ (absorbed energy for fracture versus temperature of the measurement). The definition of the reference temperature is not identical in the National Codes (Western Codes: RT_{NDT}[48], Russian Code: T_K[49]) but the absolute value for one specific steel of a specific material state is more or less very similar.

Reference temperature shift due to material degradation

Any kind of degradation processes (thermal ageing, radiation-induced embrittlement, fatigue, thermo-mechanical treatments) will cause changes of the Charpy curves ($C_v(T)$), i.e. shift of the curve toward higher temperatures and lowering of the upper-shelf energy. The shift of the reference temperature for the ductile-brittle transition (as determined from the Charpy curve) is a measure for the embrittlement due to the specific degradation process.

State-of-the-art requirements and regulations

Reference transition temperature RT_{NDT} or T_K, respectively

The definition of the reference temperature for nil ductility transition RT_{NDT} (Western Codes) or the critical temperature of brittleness T_K (Russian Code) is based on Charpy tests:

Russian Federation

The critical temperature of brittleness T_K is according to the Russian Standards [PNAE-G7-002-86][50] the temperature that fulfils the following conditions (for steels with proof stress 550-690 [MPa]):

- The mean value of the Charpy-V-notch specimen impact strength[51] at T_K must not be lower than 47 [J],
- At a temperature T_K none of three tested specimens should exhibit Charpy-V-notch specimen impact strength 70% of 47 [J]
- The mean value of the Charpy-V-notch specimen impact strength at T_K + 30 must not less than 70 [J]
- The ductile percentage of the fracture area of each specimen at T_k + 30 must not be less than 50%.

In practice this means hat T_K is the maximum value of T_{47J} and $(T_{70J} – 30)$[52]

$$T_K = max\{ T_{47J} ; (T_{70J} – 30) \} [°C].$$

United States

The ASME code defines a reference temperature for “nil ductility transition” RT_{NDT} as the temperature at 68 [J] absorbed energy (in the Charpy curve $C_v(T)$) minus 33 [°C]:

$$RT_{NDT} = T_{68J} – 33 [°C].$$

48 „reference temperature for nil ductility transition“

49 „critical temperature of brittleness“

50 p.193-198

51 Charpy test

52 In principle Finland is using the Russian Code for the Loviisa WWER-440/213 NPP. The Russian Code definition of the critical temperature of brittle fracture in the unirradiated state is: $T_{K0} = T_{47J}$

Germany

The reference temperature is defined as in the ASME Code

$$RT_{NDT} = T_{68J} – 33\ [°C]$$

France

The RCC-M Code defines the minimum values for the impact strength and the lateral expansion to be met at a given test temperature, the lowest temperature were the conditions are met is TCV which is used to determine the reference transition temperature;

$$RT_{NDT} = T_{CV} – 33\ [°C]$$

where T_{CV} is the temperature, at which each of three tested Charpy V-notch specimens exhibit at least 0,89 [mm] lateral expansion or 68 [J] absorbed energy.

There is only a slight difference between the two reference transition temperatures RT_{NDT} and T_K, namely (T_{68J}–33) – (T_{70J}–30) = T_{68J} –T_{70J}–3.

Reference temperature shift due to material degradation

The shift of the reference temperature for the ductile-brittle transition (as determined from the Charpy curve) is a measure for the embrittlement due to the any kind of degradation process. The National Codes define this shift in different ways:

Russian Federation

The critical temperature of brittleness T_k for any material state has to be determined as described above. In case of neutron embrittlement the Russian Code provides predictive formulas (see below). The critical temperature of brittleness in the unirradiated state is usually named T_{K0}.

United States

The logical assumption to use the shift of the reference temperature RT_{NDT} was not practicable, since the neutron-induced change of the upper-shelf energy of RPV steels with higher content caused upper-shelf energies in the range of 68 [J], which is the value for the definition of the reference temperature. Therefore, within the U.S. Code of Federal Regulations (Reg. Guide 1.99 rev.1 and rev.2), the shift at the absorbed energy of 41 [J] was defined as the measure for embrittlement.

Germany

The increase in transition temperature is determined from the Charpy curves at the 41 [J] level ΔT_{41J} (identical with U.S. regulations) [KTA 3201.1], [RSK Guidelines].

France

Identical to the practice in Germany.

Current plant status: Material properties of the ETE RPV steel

Workshop 2004 presentation: M. Brumovský, J. Žďárek: Reactor pressure vessel integrity (RPVI) assurance approach;

Workshop on PTS-Update 2004: M. Brumovský: Material Problems in PTS
Information from the Workshop in NRI Řež (26./27.02.2001)

In consequence of the fact that the RPV steel is Russian-type, the material data used for design and manufacture were measured according to the Russian Code regulations. The Czech Regulatory Authority has adopted these regulations. Consequently, the critical temperature of brittleness T_K is determined as described in the Russian Code.

According to the Workshop presentation, there exist the following databases of RPV material properties:

- Qualification programme: including "standard mechanical tests (tensile strength, notch toughness, hardness, bending), fracture toughness, critical temperature of brittleness tests as well as radiation resistance tests under operating temperatures up to fluences (even somewhat higher than those corresponding to the design vessel lifetime) is between 9×10^{22} and $1{,}1 \times 10^{24}$ [1/m^2], and thermal ageing tests at temperatures up to 450 [°C] and duration up to 10 000 hours."
- Extended acceptance tests: In addition to standard testing, these tests included static fracture toughness test performed on samples with a thickness of 75 mm and radiation resistance test (under three neutron fluences) for the purpose to determine the shift of transition temperature of notch toughness and static fracture toughness. The test results were included into the database of the RPV material properties.
- RPV lifetime evaluation programme: "determination of radiation damage to RPV weld joint materials at various fluences up to the design value; determination of additional fracture toughness values to increase the precision of the reference fracture toughness curve for this material in future calculations of the RPV resistance to non-ductile fracture; determination of radiation embrittlement of weld materials of both RPVs in operational conditions – temperature and fluence up to the design end-of-life fluence; determination of corrosion-mechanical properties of the base material, weld joint material and austenitic cladding in the conditions of primary coolant chemical regime".
- Surveillance specimen programme: "monitoring of changes in material properties in critical locations of RPV rings throughout the RPV lifetime under conditions close to RPV real operating conditions: specimen irradiation temperature equivalent or close to vessel wall temperature and lead factor below 2; determination of changes in properties of base material and weld joints using static tensile tests, static and dynamic fracture toughness and notch toughness tests."

During a former Workshop in NRI Řež (26./27.02.2001) the Austrian Experts had the possibility to see the protocols of the evaluation of the Charpy V-notch experiments for the determination of the critical temperature of brittleness T_{k0} of the unirradiated set of samples of the surveillance programme and part of the RPV lifetime evaluation programme [VACEK 1996] and to discuss the materials characteristics that are included in POSAR.

- Weld no. 4 was welded with two different electrode heat charges (Sv 12Ch2N2MAA, heat number 17084 and 170007) for both heat numbers surveillance samples were fabricated; the surveillance programme of ETE-1 is performed using the samples welded with the same electrode heat than weld no. 3 (T_{k0} = –50 [°C]). The other weld metal with T_{k0} = –30 [°C] will be irradiated within the surveillance programme of ETE-2. In view of the Austrian Experts Team, this is a shortcoming because the results on irradiation embrittlement of the weld material with the highest T_{k0} of ETE-1 will not be available without significant delay.
- The protocols of the T_{k0} determination were presented for review, the experimental values of –50 [°C] (weld 3) and –30 [°C] (weld 4) are realistic.
- According to the reactor passport for ETE-2 for the heat affected zone (HAZ) an initial critical temperature of brittleness is T_{k0} = –10 [°C] was measured. Considering the known experimental difficulties to determine a defined HAZ material's characteristic, for the time being the HAZ might not have to be declared the leading material with respect to embrittlement, but the future surveillance data should be investigated very carefully and with specific emphasis.

- POSAR 53168: evaluation of radiation experiments, report [VACEK 1996]: „following the state of the material during the operation using surveillance samples it will be necessary to pay maximum attention to for weld SN3 of unit 1 the were the projected A_F values from the normative codes were reached or surmounted".

  The Report [VACEK 1996] was provided for review: The results of the test reactor irradiation of the WWER-1000 materials showed that the observed embrittlement coefficients A_F were below the specified value of the Russian Code (A_F = 23 for base metal, 20 for weld) for low neutron fluences but reached or surmounted the specified values at higher fluence (A_F = 19,5 and 23). From the Austrian point of view, these results confirm that the Russian Code specifications cannot be considered a conservative basis for the structural integrity assessment.
- The discussion of the used heat treatment parameters for the RPV and the surveillance samples indicated that the heat treatment temperature and the holding times were lower for the RPV in comparison with those of the surveillance samples. This means that the samples have to be expected to show higher toughness values compared to the RPV materials.

According to [BRUMOVSKY 2004c] the VERLIFE methodology states for the determination of T_{k0}: *"If the experimentally determined values of T_{k0} from component acceptance tests (based on component passport) are known, they can only be used in the case that the following temperature margin δT_M will be added; the margin has to take into account the scatter of the values of mechanical properties in the semi-product."*

"δT_M is the mean quadratic deviation of T_{k0} determined for the given semi-product in the frame of qualification test or in the frame of a set of identical semi-products established during production of the component by the identical technology. If this value is not available, the application of the following values is suggested:

δT_M = 10 [°C] for the base material,

δT_M = 16 [°C] for the weld metals."

Evaluation

The Czech determination of the critical temperature of brittleness T_k is defined and performed in accordance with the Russian Code regulations, which are quite close to the Western practice. The determination of the shift of this temperature caused by neutron embrittlement is also performed and defined in accordance with the Russian Code regulations, at least as long as the T_K is used within the RPVI assessment. The Russian Code defines an embrittlement factor AF which is specified for WWER-1000 RPV steels with the values A_F = 20 for welds and A_F = 23 for base metal.

Material characteristic	Russian Fed.	United States	Germany	France	Czech Rep.
Reference temperature	min {T_{47J};(T_{70J} – 30)}	$T_{68\,J}$ – 33 [°C]	$T_{68\,J}$ – 33 [°C]	T_{cv} – 33 [°C]	min {T_{47J};(T_{70J} – 30)}
ΔT_{RTNDT} or ΔT_k	Code specification	$\Delta T_{41\,J}$	$\Delta T_{41\,J}$	$\Delta T_{41\,J}$	Code specification

Database of material characteristics:

The Austrian Experts were not given the opportunity to review the material characteristics derived within the qualification programme and the extended acceptance test.

During former Workshops (2000, 2001) Austrian Experts' Teams of different composition had the possibility to review the evaluation of the Charpy curve with respect to the determination of the critical temperature of brittleness in the initial unirradiated state, performed as part of the RPV lifetime evaluation programme. This evaluation is in accordance with the current practice.

The Austrian Experts' Team was not informed about the requirements in the context of VERLIFE back in 2000/2001, in between other it was therefore not checked, whether the defined temperature margin δT_M (10 [K] for the base material and 16 K] for the weld metals) has been taken into account.

The values for the initial critical temperature of brittleness T_{k0} known to Austrian Experts from the POSAR are as follows:

Reactor	Material	State	T_{k0} [°C]
ETE-1	15Ch2NMFA-A	base (5)	–50
ETE-1	15Ch2NMFA-A	base (6)	–60
ETE-1	Sv-12Ch2N2MAA+FC-16A	weld 3	–50
ETE-1	Sv-12Ch2N2MAA+FC-16A	weld 4	–30
ETE-2	Sv-12Ch2N2MAA+FC-16A	weld 3	–30
ETE-2	Sv-12Ch2N2MAA+FC-16A	weld 4	–20
ETE-2		HAZ/at weld 3	–10
ETE-1/2	No information provided	Weld 2	

Since the highest values (apart from HAZ, see below) for the initial critical temperature of brittleness T_{k0} are in weld no.4 of unit 2 this material should be considered leading with respect to embrittlement, although the neutron flux at weld no. 4 is about 80% compared to weld no. 3. The surveillance program covering this weld metal will only be performed during ETE-2 operation [ŘEZ 2001].

Although the initial critical temperature of brittleness T_{k0} of HAZ/ETE-2 is the highest T_{k0} value, for the time being the HAZ might not have to be declared to be the leading material with respect to embrittlement because of the known experimental difficulties to determine a defined HAZ material's characteristic. The future surveillance data should be investigated very carefully and with specific emphasis.

During the Workshop in Prague the data for the initial critical temperature of brittleness T_{k0} presented in [PISTORA 2004b] differ from the above values with respect to weld 4/unit-1:

Reactor	Material	State	T_{k0} [°C]
ETE-1	15Ch2NMFA-A	base	–50
ETE-1	Sv-12Ch2N2MAA+FC-16A	weld 3	–50
ETE-1	Sv-12Ch2N2MAA+FC-16A	weld 4	–60
ETE-2	15Ch2NMFA-A	base	–60
ETE-2	Sv-12Ch2N2MAA+FC-16A	weld 3	–30
ETE-2	Sv-12Ch2N2MAA+FC-16A	weld 4	–20
ETE-1/2	no information provided	weld 2	

Welding and weldment performance of weld no. 4 at ETE-1 were discussed during the Workshop in Řež 25./26.02.2001: Weld no. 4 was welded with two different electrode heat charges (heat number 17084 and 170007), for both heat numbers surveillance samples were fabricated; the surveillance programme of ETE-1 is performed using the samples welded with the same electrode heat than weld no. 3 (T_{k0} = –50°C). The other weld metal with T_{k0} = –30°C will be irradiated within the surveillance programme of ETE-2.

It is recommended that the results of the surveillance samples irradiated in unit 2 (esp. specimens of weld no.4/unit-1 and weld no.4/unit2, including HAZ) should be monitored with special emphasis during the next years. At the same time it would be desirable to obtain information whether specimen of weld number 2 are included in the PTS considerations.

4.2.2 Verification of RPV material specimen neutron irradiation embrittlement

Description of the issue – fundamentals

Reactor materials are exposed to the radiation resulting from fission processes in the reactor core, mainly neutrons and gamma rays.

Due to the neutron impact the crystal structure of the steel is disturbed: lattice defects are generated in displacement cascades by high-energy recoil atoms from neutron scattering and neutron reactions.

The resulting point defects (individual interstitials, vacancies, Frenkel defects = interstitial-vacancy pairs) may interact, agglomerate to clusters, diffuse through the material, annihilate, etc. The diffusion of the primary defect can also lead to enhanced solute diffusion (radiation enhanced diffusion RED) and the formation of defect-solute cluster complexes, solute clusters, and distinct new phases (precipitates).

This nanoscale inhomogeneous structure causes dislocation pinning and pile-ups, which results in hardening effects of the material (increase of the yield stress σ_y).

Cleavage (brittle fracture) occurs, when the stress concentration at a notch or crack tip exceeds a critical stress over a micro-structurally significant length. Steels behave brittle at low temperature; there exists a ductile-brittle transition temperature defining the change of brittle to tough behaviour.

<u>Irradiation embrittlement nanostructural effects</u>

Irradiation induced segregation of impurities at grain boundaries may decrease the critical stress for cleavage initiation and hence increasing the ductile-brittle transition temperature (ΔT_t). This effect is called radiation-induced embrittlement. Micromechanical models are consistent with the experimentally observed relation between radiation-induced increases of ductile-brittle transition temperature and yield stress $\Delta T_t/\Delta\sigma_y \approx 0{,}6\pm0{,}2$ [K/MPa]; [ODETTE 2001].

The current understanding of embrittlement nano-features is based on micro structural and micro chemical characterizations (small angle X-ray and neutron scattering, electron microscopy, atom-probe field ion microscopy, positron annihilation spectroscopy) and thermodynamic-kinetic models using molecular dynamics and Monte Carlo computer simulations. These nano-features can be divided in three categories [ODETTE 2001]:

- Copper rich or catalysed manganese-nickel rich precipitates (CRPs/MNPs).
- Unstable matrix defects (UMD) that form cascades even in steels with low or no copper but then dissolve in relatively short times; UMDs are believed to be sub-nm vacancy clusters, complexed with solutes, UMDs may serve as nucleation sites for larger SMFs.
- Stable matrix features (SMF) that persist or grow under irradiation even in steels with low or no copper; SMFs range from loose aggregates of solutes to nanoscale precipitates.

In sensitive steels with copper contents greater than about 0,05 ÷ 0,1% the CRPs are considered as the dominant hardening mechanism. The CRP based hardening saturates at high neutron fluences due to copper deletion in the matrix. At very high neutron fluxes (for instance in test reactor irradiations) the population of UMDs becomes significant, acting as vacancy-interstitial sink, consequently reducing radiation induced diffusion and the CRP evolution. At very low neutron fluxes the CRP evolution may be accelerated due to the contribution of radiation enhanced copper diffusion. This so-called dose rate effect is problematic in case predictive embrittlement tests using test reactor irradiation, since the results of accelerated irradiation (high neutron fluxes compared to the real neutron flux at the RPV wall) can be strongly non-conservative.

Nickel and manganese strongly bind and enhance the effect of copper, thus increasing the volume of the precipitates and in consequence the hardening and embrittlement of the steel. In some cases this can result in manganese-nickel-rich precipitates with a small copper core. Up to intermediate neutron fluence levels, pure MNPs have not been observed by experimental means. More recent research results indicate that with high nickel and at the same time high manganese contents (Mn > 0,8% for the WWER 1000-320 RPV steel) the MNP formation becomes a dominant factor for embrittlement (also [BRUMOVSKY 2004c]).

Research work on embrittlement of the WWER materials with special emphasis on base metal and HAZ with respect to grain boundary precipitations is on the way [ENGLISH 2003].

Radiation embrittlement due to the operation of the plant is dependent on many factors, such as the initial material properties (material state, i.e. base metal or weld; composition, esp. impurity levels, heat treatments), the operational temperature at the vessel wall, neutron spectra and neutron flux at the vessel wall. Due to the strong dependence on the impurity level (mainly copper and phosphorus) and the alloying element Ni it is important to have a quantitative knowledge on the compositional dependence of the embrittlement in order to be able to predict the degradation of the vessel steel during operation and to estimate the safety margin throughout the service life of an individual RPV.

Material data on the brittle/tough behaviour are usually derived from Charpy tests. Only in the last decade, the direct experimental determination of fracture toughness using fracture mechanical specimens has become important. For older power plants, there are not enough fracture toughness data for the virgin (unirradiated) RPV materials and not enough representative sample reserves for an appropriate surveillance program. Due to the scatter of such data, a rather vast entity of samples would be needed in order to get reliable results.

The PTS analysis requires fracture toughness curves (fracture toughness versus temperature) as characteristics for the material state to be compared with each calculated loading path for a specific accident transient and a specific postulated crack. The fracture toughness curve for the RPV steel during operational life is dependent on the irradiation embrittlement, i.e. the curve is shifted with increasing neutron fluence toward higher temperatures. As described above the lack of sufficient “real” fracture toughness data has forced the Regulatory Authorities to define a method to describe this shift. According to the ASME Code the shift of the reference transition temperature, as derived from Charpy measurements ΔT_{41J} has to be used.

State-of-the-art requirements and regulations

Neutron irradiation induced embrittlement causes a shift of the Charpy curve to higher temperatures and a lowering of the upper shelf energy. For the PTS analysis following Western Standards, it is of importance to know the fluence dependence of the reference transition temperature shift. There are usually two possibilities: either measured data of irradiated representative RPV materials samples (surveillance program) with acceptable credibility or the use of predictive formulas, if foreseen within the National Regulatory Guidelines (all formulas assume that the irradiation temperature is 288 ÷ 290 [°C]).

Russian Federation

The Russian Code specifies the shift of the reference transition temperature T_K using an empirical formula that covers the effect of Cu and P impurities. In [PNAE-G-7-002-86] the analytical formula for the fluence dependence is given:

$$\Delta T_F = A_F.(f/10^{22})^{1/3}$$

where f is the neutron fluence in ^1_0n /m^2, E > 0,5 [MeV].

A_F is the embrittlement coefficient, in case of WWER-1000:

$$A_F = 23 \text{ for base metal, } A_F = 20 \text{ for welds}$$

(the Cu content has to be < 0,10% for base metal, < 0,08% for welds, the P content has to be < 0,010%; and the Ni content has to be below 1,6%)

United States

When credible surveillance results are not available, the predictive formula as defined in [Reg. Guide 1.99, rev.2] is to be used:

The adjusted reference temperature

$$RT^{adj} = RT_{NDT}^{initial} + \Delta RT_{NDT} + margin$$

$$\Delta RT_{NDT} = (CF).f^{(0,28-0,1.\log f)}$$

CF .... the chemical factor, given in tabular form as a function of the Cu and Ni content for base and weld material

f.................... neutron fluence in 10^{19} [n/cm^2]], E > 1 [MeV]

The neutron fluence at any depth in the vessel wall is: $f = f_{surface}.e^{-0,24\,x}$

where x .... is the depth into the vessel measured from the inner surface (in inches)

margin.......... is added to obtain a conservative value of the adjusted reference temperature = $2.\sqrt{(\sigma_I^2 + \sigma_\Delta^2)}$

σ_I.................. is the standard deviation for the initial RT_{NDT} (precision of the test method) and

σ_Δ.......................... is the standard deviation for ΔRT_{NDT} (15,5 [K] for welds, 9,5 [K] for base metal)

In case of credible surveillance, data sets[53] the experimental values should be fitted to the above formula of ΔRT_{NDT} – fluence dependence.

Germany

ΔRT_{NDT} is given in graphical form (design curves) as a function of the Cu content and the fluence. The P content is taken into account by increasing the Cu content by 0,01 for every 0,002% P above a content of 0,012% P. Experimental values of representative surveillance samples may be used to determine the adjusted RT_{NDT}. If this experimental shift exceeds the design value, the safety of the RPV has to be demonstrated.

France

In the French Code RCC-M different formulas are used for the design state and for surveillance purposes:

Design: (Appendix ZG, ZG3430):

$$\Delta RT_{NDT} = [22 + 556.(\%Cu - 0,08) + 2778.(\%P - 0,008)]\ (f/10^{19})^{1/2},\ E < 1\ [MeV]$$

---

[53] the conditions for credibility are discussed in [Reg.Guide 1.99, rev. 2]

(in case the Cu content is lower tan 0,08 wt% and/or the P content is below 0,008 wt% the values 0,08 and/or 0,008 have to be introduced into the formula)

The different predictive formulas, tables and graphs in the National Codes are resulting from statistical evaluations of all available data on the RPV steels used. The observed strong embrittlement of US steels has been attributed to the relatively high Cu content in combination with the Ni content. These experiences have been adopted more or less by the German regulations. In consequence of the observed high neutron embrittlement in the NPPs Obrigheim and Stade the RPV steels were optimized with respect to the purity (esp. minimization of the Cu content). The French utilities were obviously never confronted with exceeding embrittlement of the RPV steel, probably due to higher purity of the steel.

Russian experiences of strong embrittlement of the RPV steel in WWER-440/230 reactors were explained by the rather high phosphorus impurity content in combination with the Cu-content.

The regulatory requirements therefore reflect the national problems of the specific steels used in their nuclear industry. National requirements may differ significantly since they are based on the statistical evaluation of the respective surveillance results; there is no "universal law" for the different RPV steels [GERARD 2003a].

Current plant status: Neutron embrittlement of Temelín RPV materials

Workshop 2004 presentation: M. Brumovský, J. Žďárek: Reactor pressure vessel integrity (RPVI) assurance approach, Workshop on PTS-Update 2004: M. Brumovský: Material Problems in PTS

According to the cited presentations several programmes include experimental data on the neutron embrittlement of the RPV steel that have been performed before reactor operation in order to confirm the specifications of the Russian Code (the surveillance programme is performed during the plant operation for monitoring the embrittlement):

- Qualification programme: "radiation resistance tests under operating temperatures up to fluences (even somewhat higher than those corresponding to the design vessel lifetime) between 9×10^{22} and $1,1 \times 10^{24}$ [n/m^2], and thermal ageing tests at temperatures up to 450 [°C] and duration up to 10 000 hours."
- Extended acceptance tests: "radiation resistance test (under three neutron fluences) for the purpose to determine the shift of transition temperature of notch toughness and static fracture toughness. The test results were included into the database of the RPV material properties."
- RPV lifetime evaluation programme: "determination of radiation damage to RPV weld joint materials at various fluences up to the design value; determination of additional fracture toughness values to increase the precision of the reference fracture toughness curve for this material in future calculations of the RPV resistance to non-ductile fracture; determination of radiation embrittlement of weld materials of both RPVs in operational conditions – temperature and fluence up to the design end-of-life fluence; determination of corrosion-mechanical properties of the base material, weld joint material and austenitic cladding in the conditions of primary coolant chemical regime" – all results are better than predicted by Soviet Code.

According to the information given during the PTS-Update Workshop [BRUMOVSKY 2004c], the VERLIFE methodology includes regulations with respect to the determination of the experimental plant-specific neutron induced shift of the critical temperature of embrittlement:

- *"The experimentally derived shifts of transient temperatures must be based on at least three different neutron fluences."*

- *"The mean trend curve should be vertically shifted upward by the value of δT_M. If any experimental point exceeds this adjusted trend curve, the curve should be shifted further until it bounds all data. This upper boundary of the shifts is to be used in assessment of RPV resistance against fast fracture."*
- *"It is not allowed to extrapolate shifts of the transient temperatures for fluences higher than 2-multiple of the maximum fluence for the experiment."*

Evaluation

The observed high radiation embrittlement in the RPV steel of the WWER-440 reactors is assumed to be due to the relatively high copper and phosphorus impurity levels. Steel for the manufacture of the reactor pressure vessel for WWER-1000 reactors has therefore a decreased copper and phosphorus content. Because of the greater design dimensions (wall thickness, ring diameter), compared to the WWER-440 RPV the steel was alloyed with increased nickel content (1,1 to 1,9%) in order to enhance the manufacturability.

The predictive values for neutron embrittlement in the Russian Code are based on experimental test results on the steel 15Ch2MNFAA, irradiated 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 Řež using original RPV materials of the NPP Temelín unit-1 have also been realized with high lead factors (about 160 or higher). This means that a possible dose rate effect (higher embrittlement at lower neutron flux compared to high neutron flux for identical fluence) could have falsified the results.[54]

The restricted validity of the Russian Code specifications concerning the embrittlement of the WWER-1000 RPV steel can also be found in the IAEA Guidelines for the pressurized thermal shock analysis [IAEA 1997]: the European experts stated with respect to the WWER-1000 materials, that the normative embrittlement coefficient values of 20 and 23 <u>are not conservative</u>, if the nickel content of the steel is above 1,3%.

The nickel content is also restricted according to other national regulations:

KTA 3201.1: the maximum allowed content is 0,85 wt% Ni,

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 should be considered non-conservative ([KRYUKOV 1997], [KRYUKOV 1999], [KRYUKOV 2000], [GRYNIK 1999], [VIEHRIG 1999].

Experimental results indicate that the nickel content of the RPV steel is causing enhanced embrittlement. In NPP Temelín the Ni content in the RPV steel is 1,63 ÷ 1,66%. For comparison the Ni content of other WWER-1000 plants [KRYUKOV 1999]:

[54] A question of the Austrian Experts during the Workshop 2004 on the to possible influence of the dose rate effect was answered by J. Žďárek in the sense that this effect cannot be excluded and will be studied in future research programmes

Table 1: Ni contents of welds in WWER-1000 plants

NPP	BOL	Weld nr.	Ni content (%)
South Ukraine-2	1984	3	1,77
South Ukraine-2	1984	4	1,74
South Ukraine-3	1989	3	1,72
South Ukraine-3	1989	4	1,72
Rovno-3	1986	3	1,64
Rovno-3	1986	3	1,59
Saporoshje-3	1986	3	1,55
Saporoshje-3	1986	3	1,57
Saporoshje-4	1987	3	1,70
Saporoshje-4	1987	4	1,70
Saporoshje-5	1989	3	1,60
Saporoshje-5	1989	4	1,60
Khmelnitzki-1	1987	3	1,88
Khmelnitzki-1	1987	4	1,88

Since the nickel content of the weld metal of ETE-Unit 1 is in the range 1,63 ÷ 1,66% the specified values in the Russian Code cannot be considered conservative.

Since the first Temelín specific surveillance programme data of irradiated samples will be available in 2005, the published results on WWER-1000 surveillance programmes will be analysed. The experimental data that are the basis for the Russian Code specifications were mainly determined with irradiation in test reactors with high lead factor. Due to the high lead factors the existence of a dose rate effect[55] might falsify the experimental results, this could be the reason for the reported non-conservatism the specified embrittlement [GYRNIK 1999], [KAMENOVA 1999], [KRYUKOV 1997], [KRYUKOV 2000], [KRYUKOV 2000a].

<u>Review on published surveillance results:</u>

Published data on WWER-1000 material irradiation have been reviewed in order to have some indications on the experimentally observed embrittlement in comparison with the specification in the Russian code. According to the Russian Code the embrittlement factor is 20 for welds and 23 for base metal.

In [KAMENOVA 1999] the experimentally determined A_F-values for weld metal surveillance samples from different WWER 1000 plants are found to be between A_F = 6 and 29. Other authors [KRYUKOV 2000a], [BOEHMERT 2000] report even higher embrittlement coefficients. In the table below the published A_F values and some information on the chemistry are compiled. It is interesting to note, that the lowest A_F values do not coincide with the lowest Ni-content, neither with the lowest P or Cu content.

[55] lower embrittlement at high neutron dose rate (flux) compared to irradiation with lower neutron flux for identical irradiation dose.

Table 2: Surveillance samples of WWER-1000 RPV weld metal

Reference	NPP/Unit		Ni [%]	P [%]	Cu [%]	A_F	adjusted $A_F^{290°C}$
[Kamenova 99]	Balakovo-1	weld	1,88	0,009	0,028	29	**35**
[Kamenova 99]	Kalinin-1	weld	1,76	0,01	0,04	35	**41**
[Kamenova 99]	Novovoronesh-5	weld	1,21	0,014	0,04	8	**14**
[Kamenova 99]	S Ukraine-1	weld	1,72	0,008	0,05	16	**22**
[Kamenova 99]	S Ukraine-2	weld	1,72	0,005	0,06	6	**12**
[Kamenova 99]	Kozloduy	weld	1,7	0,009	0,03	12	**18**
[Boehmert 2000]	Archive/255°C	weld	1,71	0,04	0,012	47,5	**33,5**
[Kryukov 2000a]	Unit 1	weld	1,88	0,009	0,028	32	**38**
[Kryukov 2000a]	Unit 2	weld	1,76	0,010	0,040	37	**43**
[Kryukov 2000a]	Unit 3	weld	1,21	0,014	0,040	10	**16**

The irradiation temperatures for surveillance-samples in WWER-1000 pressure vessels were according to [KAMENOVA 1999] "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] for the adjustment with respect to the irradiation temperature

$$A_F(T_{irr}) = A_F(T_v) + K.x.(T_v - T_{irr})$$

(K = 0,2 for base metal and 0,4 for weld metal)

the respective adjusted A_F value (adjusted for 290 [°C]) can be calculated from data of another irradiation temperature T_{irr}; these values are given in the last column of table above.

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 RPV wall and this could introduce a lack of conservatism". Considering this irradiation temperature of 320 [°C] an adjustment would further increase the A_F values by +6.

WWER-weld metal irradiated at 255°[C] in Rheinsberg showed A_F = 47,5; the adjustment to the vessel temperature of 290 [°C] results in $A_F^{290°C}$ = 33,5 (see table 1, last row)[56] .

In [KRYUKOV 2000a], surveillance data (the NPPs are not named in the publication) were investigated with Ni content of the weld between 1,21% and 1,88%, and of the base metal between 1,13% and 1,35%. The determined embrittlement coefficients of the welds were found to depend on the Ni and Mn content, which was not observed for forgings and heat affected zone (HAZ) materials.

The complexity of the alloying influence on A_F values is apparently also a consequence of the combined effect of nickel and manganese. A content of more than 0,8 [% wt.] manganese is most likely to be an essential contributor to irradiation embrittlement. For the Temelín RPVs however the manganese contents are limited to a maximum at 0,49 [% wt.] for the base material and at 0,7 [% wt.] for the weldments [BRUMOVSKY 2004c].

[56] Temperature uncertainties of □5°[C] change the AF value by □2 (higher temperature ⇒ lower AF)

Two TACIS projects have been focused on the WWER-1000 RPV problems [KRYUKOV 2000]:

- *TACIS-92 Evaluation of reactor pressure vessel embrittlement of South Ukraine NPP including embrittlement aspects.*
- *TACIS-94 Integrity assessment of the WWER-1000 RPVs including embrittlement aspects.*

"In the framework of these projects, the validity and representatively of WWER-1000 surveillance data and other experimental results have been done. But due to the low fluence value and insufficient number of surveillance specimens the accuracy of radiation embrittlement assessment of RPVs was not high. It was also confirmed that the specimen temperature was possibly higher than the vessel wall temperature. In this case the surveillance results for vessel embrittlement assessment may give non-conservative forecast."

Summarising these facts it can be stated that the evaluation of published surveillance results from WWER-1000 materials, taking into account the estimated irradiation temperatures, does show the strong uncertainties about the neutron embrittlement of WWER-1000 steel. It is obvious in the first place that the specification in the Russian Code (A_F = 20 for welds, A_F = 23 for base material) cannot be considered conservative.

Although the first reliable results will be available from the Temelín surveillance program, these results cannot eliminate the uncertainties on the WWER-1000 RPV steel embrittlement: the RPV specific surveillance program cannot provide a reliable statistical background for the prediction of the material degradation, since every set of withdrawn and evaluated samples gives one single data point in the embrittlement versus irradiation time plot.

The first capsule with irradiated samples has been withdrawn during May 2004; the evaluated data will be available one year thereafter.

According to the requirements of the VERLIFE methodology [BRUMOVSKY 2004c] the experimental assessment of the neutron embrittlement is required to include temperature margins; at the same time it is required to use upper bound curves in case these margins would not cover all experimental data. The extrapolation of the T_k shifts for fluences higher than twice the maximum experimentally covered fluence is not allowed.

The Austrian Experts' Team recommends monitoring in the future whether these requirements are met with in the PTS updating by the operator.

In that context, it is interesting to note that a warning was posted in POSAR, page 5.3-18 of unit 1 [POSAR]:

"Pursuing the RPV materials condition during operation with the surveillance program it will be necessary to pay maximum attention to the ***mechanical properties degradation of the weld metal*** *(weld no 3, RPV Unit-1)."*[57]

It has to be stated that there are indications – not only based on Russian research results, but also on ETE results – that the specified embrittlement coefficients A_F cannot be considered to be conservative. There are also indications that the initial critical temperature of brittleness T_{k0} can vary by tens of degrees from one weld metal charge to another. Both uncertainties should be taken into account.

The initial critical temperatures of brittleness T_{k0} of the leading materials in the RPVs of ETE-1 and ETE-2 have been summarized in 4.2.1:

[57] Accentuation by the authors

Table 3: Material Code and initial critical temperatures of brittleness: RPV material at ETE1 and ETE2

Reactor	Material	State	T_{k0}(°C)
ETE-1	15Ch2NMFAA	base (5)	–50
ETE-1	15Ch2NMFAA	base (6)	–60
ETE-1	Sv-12Ch2N2MAA+FC-16A	weld 3	–50
ETE-1	Sv-12Ch2N2MAA+FC-16A	weld 4	–30
ETE-2	Sv-12Ch2N2MAA+FC-16A	weld 4	–20
ETE-2		HAZ	–10

Using the Russian formula for the prediction of the embrittlement (increase of T_k) with irradiation fluence (time of operation) the T_k values of these materials were calculated for 1, 5, 10, 15, 20 and 40 years of operation (see the next table). For the welds no. 4 it is taken into account that the neutron flux is lower than in the belt region (about 80% of the flux in the center line).

The Russian Code and the IAEA Guidelines [IAEA 1997] consider the uncertainties of the T_k values by a safety factor of ΔT = 10 [K]. The U.S. regulations require the use of a safety margin to cover the uncertainties of the experimental method for the determination of the initial RT_{NDT} and the uncertainties of the determination of ΔT_{RTNDT} (15,5 [K] for welds and 9,5 [K]). Other National Codes do not provide rules for the use of safety margins to consider the uncertainties. The Czech VERLIFE methodology application does not include any safety factor (see also under "evaluation of VERLIFE Methodology" 3.1.4).

In the following table the calculated critical temperatures of brittleness T_k ("net") and these values plus ΔT = 10 [K] ("incl. ΔT") are summarised for distinct years of operation.

Table 4: Critical temperature of brittleness values for base and weld metal at ETE-1 and ETE-2

Years of operation	Weld 3/ETE-1		Base/ETE-1		Weld 4/ETE-1		Weld 4/ETE-2		HAZ/ETE-2	
	A_F°= 20		A_F°= 23		A_F°=20/80%F[58]		A= 20/80%F		A_F°= 23/80%F	
	net	incl. ΔT	net	incl. ΔT	net	incl. ΔT	net	incl. ΔT	net	incl. ΔT
0	–50	**–40**	–50	**–40**	–30	**–20**	–20	**–10**	–10	**0**
1	–28	**–18**	–24	**–14**	–9	**1**	1	**11**	14	**24**
5	–12	–2	–6	4	6	16	16	26	31	41
10	–2	8	6	16	15	25	25	35	42	52
15	5	15	14	24	21	31	31	41	49	59
20	11	21	20	30	27	37	37	47	55	65
40	27	37	38	48	41	51	51	61	72	82
[a]	**[°C] [K]**									

The so-called embrittlement curves (critical temperature of brittleness as a function of the time of operation) as predicted using the specified embrittlement coefficients of the Russian Code are compiled in the following figure for the different materials in the RPVs (unit 1 and unit 2) together with the maximum allowable critical temperature T_k^a for the accident transient C2 taking into account the safety factors as required by the IAEA Guidelines and applying the WPS criterion at the 80% level (see chapter 3.6).

[58] F: fluence in 10^{22} n/m^2

As has been reported, the specified embrittlement coefficients for WWER-1000 materials in the Russian Code cannot be considered conservative. In the next figure the embrittlement curves have been re-calculated for only slightly increased embrittlement factors: A_F = 25 for the weld material (Russian Code: 20) and 28 for the base material (Russian Code: 23). It can be seen that this relatively small change of the A_F value causes a significant aggravation of the critical embrittlement already in the first 5 years of operation.

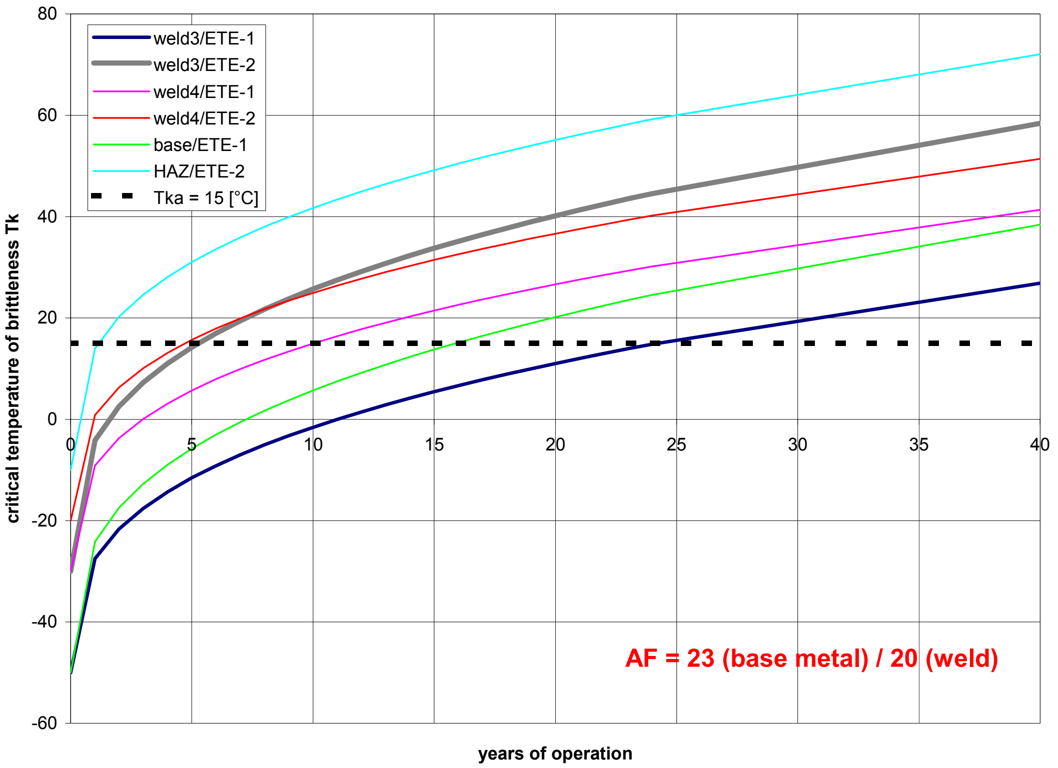

Figure 4: Neutron induced embrittlement of ETE materials assuming of the specified embrittlement coefficients as defined in the Russian Code for WWER-1000 materials: A_F = 23 for the base metal, A_F = 20 for welds

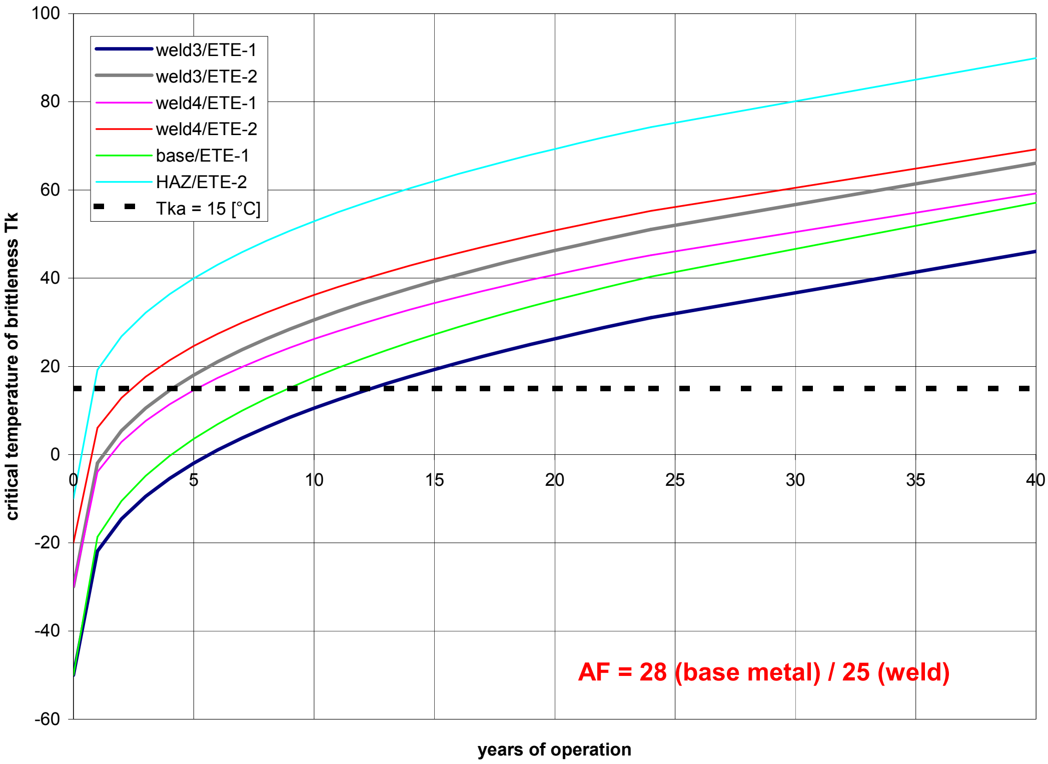

Figure 5: Neutron induced embrittlement of ETE materials assuming embrittlement coefficients higher than the specified values in the Russian Code for WWER-1000 materials: A_F 28 for the base metal, A_F = 25 for welds

It has to be kept in mind that the uncertainties about the true embrittlement of the WWER-1000 materials that will not be eliminated by the first sets of surveillance.

4.2.3 Fracture toughness curves from RPV material specimen– RPV lifetime evaluation

Description of the issue – fundamentals

The assessment of the structural integrity or the residual lifetime of an RPV by a PTS analysis includes the comparison of the calculated load path in case of a PTS transient and the actual material state of the RPV steel which degrades mainly due to neutron embrittlement. This material state is described by the fracture toughness as a function of temperature K_{Ic} $(T-T_k)$. Since most surveillance programmes included mainly Charpy specimens the relevant fracture toughness curve has to be determined by empirically deduced formulas using the material characteristics derived experimentally from Charpy curves: RT_{NDT} or T_K, respectively. The values ΔT_{41J} or ΔT_K, respectively are used to derive the fracture toughness curve K_{Ic} $(T-T_k)$ for the embrittled material state by shifting it.

In the last few years the so-called “Master curve methodology” has been elaborated, which is based on direct experimentally derived fracture toughness characteristics. Since most of the utilities operated today do not have enough representative specimens (unirradiated and irradiated) for direct fracture toughness measurements, the Master curve concept is not yet common use. There are tendencies to implement the methodology into the National Codes but it seems that the Master Curve concept will only be accepted in a few years for RPV materials ageing assessment [GERARD 2003a].

The major open technical issues are the application of the Master Curve concept outside the temperature region $-50<T-T_0<+50$ [°C], and the applicability of the master curve for materials failing by grain boundary fracture. In addition, the use of the Master Curve for low constraint geometries in elastic-plastic loading needs still refinement. Other open issues are related to the application of the Master Curve concept within the context of the present rules [GERARD 2003b].

Another problem for the application of the master curve is the relatively small fracture toughness database. Approved trend curves for the Codes need to be based on a relatively large database; to provide such a database with surveillance samples will need tens of years [VALO 2003]. The development of the fracture toughness database is topic of the programme FRAME [VALO 2003].

State-of-the-art requirements and regulations

These curves are generally defined in the National Codes or Regulations using the determined reference temperature RT_{NDT} or T_K for the actual material state:

<u>Russian Federation:</u>

for WWER-1000 base materials

- Normal conditions: $K_I = 37+5,5 \exp [0,0385. (T-T_K)]$ [MPa.m$^{1/2}$]
- Emergency conditions: $K_I = 74 + 11 \exp[0,0385.(T-T_K)]$ [MPa.m$^{1/2}$]

for WWER-1000 weld materials:

- normal conditions: $K_I = 17,5+26,5 \exp[0,0217.(T-T_K)]$ [MPa.m$^{1/2}$]
- emergency conditions: $K_I = 35 + 53 \exp[0,0217.(T-T_K)]$ [MPa.m$^{1/2}$]

<u>United States</u>

$$K_{Ic} = 36,48 +22,783 \exp [0,036.(T-RT_{NDT})]\ [MPa.m^{1/2}].$$

<u>France</u>

$$K_{IC} = \min \{36,5 + 3,1 \exp [0,036.(T-RT_{NDT} +55,5)];\ 220\}\ [MPa.m^{1/2}]$$

(RCC-M, ZG3420); this formula may be transformed into:

$$K_{IC} = 36,5 +22,86 \exp [0,036.(T-RT_{NDT})]\ [MPa.m^{1/2}]$$

The graphic representation of these data is more or less identical with the ASME curve.

<u>Germany</u>

No analytical expression was given in the original Code rules; the version of 06/96 has adopted the ASME formula that was always used in practice.

<u>IAEA</u>

The IAEA Guidelines [IAEA 1997] recommends the use of the following fracture toughness curve:

$$K_{IC} = \min \{26 + 36 \exp [0,02.(T-TK)];\ 200\}\ [MPa.m^{1/2}]$$

United Kingdom

The U.K. regulation is essentially non-prescriptive and does not define fracture toughness curves. There are two “Safety analysis Principles” dealing with fracture toughness requirements [UK Reg. 1994]:

Principle 149: *“a metal pressure retaining body should, were appropriate, have design characteristics which prevent fast propagation of any defect. Design and conditions in which components of the coolant pressure boundary could exhibit brittle fracture behaviour should be avoided.”*

Principle 161: *“for metal pressure vessels and circuits the operating regime should ensure that they display ductile behaviour when significantly stressed”.*

Current plant state: Fracture toughness curve in the VERLIFE methodology

Workshop presentation: M. Brumovský “Unified procedure for lifetime assessment of components and piping in WWER type NPPs”

The formula for the fracture toughness curve as defined within the VERLIFE methodology is:

For the Master Curve approach:

$$K_{Ic}(med) = 30+70.exp[0,019.(T-T_0)]$$

$$K_{Ic}(5\%) = 25,2+36,6.exp[0,019.(T-T_0)]$$

$$K_{Ic}(95\%) = 34,5+101,3.exp[0,019.(T-T_0)]$$

In case of application of transition temperatures derived from the Charpy curve (critical temperature of brittleness T_k) the following formula is to be used:

$$K_{Ic} = 26 + 36.exp[0,02.(T - T_K)]\ [MPa.m^{1/2}]\ [PISTORA\ 2002];\ [PISTORA\ 2004a]$$

Evaluation

The comparison of the cited fracture toughness curve $K_{Ic}(T-T_0)$ formulas shows that the fracture toughness curves to be used in France and Germany are practically identical to the ASME curve.

The formula for the fracture toughness curve used in the VERLIFE methodology is identical to the formula given in the IAEA Guidelines. This curve is the most conservative one above about 70 [$MPa.m^{1/2}$]. Below this value the ASME curve is slightly more conservative (see the following figure).

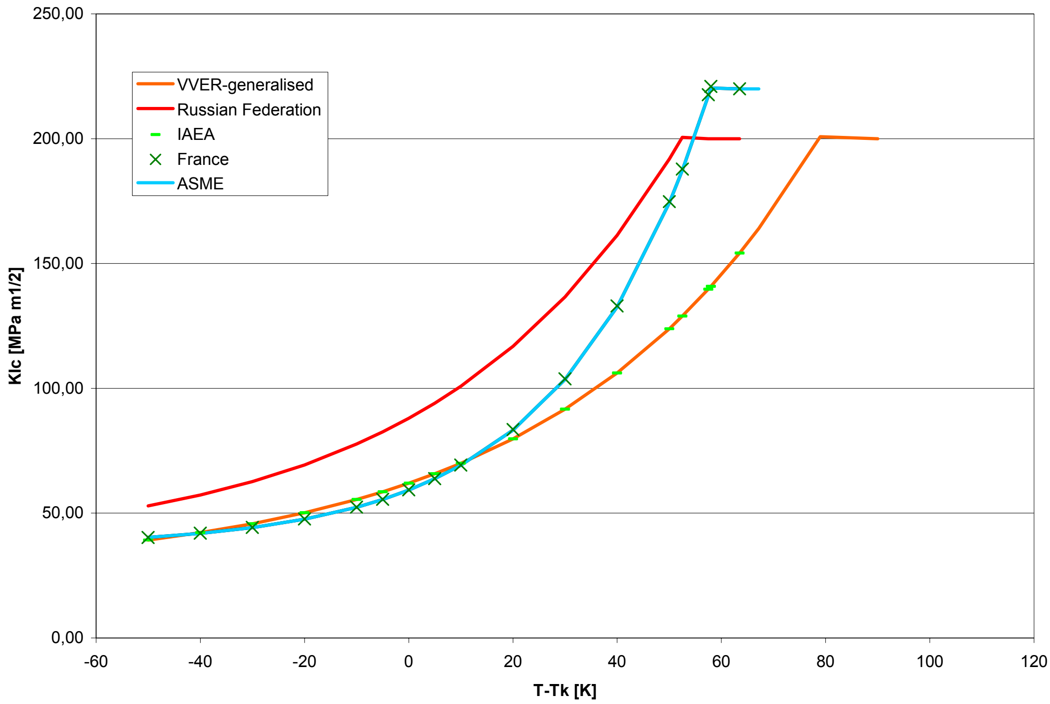

Figure 6: Comparison of fracture toughness curves ("VVER generalised" is the formula used in VERLIFE)

The demonstration of the conservatism of the used fracture toughness curve KIc(5%)(T-T0) is using experimental static fracture toughness data only from WWER-440 base and weld materials.

Nevertheless, the experimental static fracture toughness data of surveillance materials (also only WWER-440 materials) as presented during the Workshop [BRUMOVSKY 2004b] clearly show that the scatter of the irradiated materials is very high so that not even the K_{Ic}(5%) can be considered a lower bound envelope. Therefore, it was not demonstrated that the fracture toughness curve as used could be considered conservative for irradiated WWER-440 materials (see also [BRUMOVSKY 2003a, BRUMOVSKY 2003b, BRUMOVSKY 2003c]).

There is also no evidence at all that the static fracture toughness data from WWER-1000 materials will be described conservatively by the fracture toughness curve used.

4.3 Conclusions concerning surveillance programme – material embrittlement

The following conclusions concerning the surveillance programme in the NPP Temelín and the material embrittlement of ETE RPV materials can be summarized:

- The modified surveillance programme in the NPP Temelín allows the determination of reliable embrittlement data with respect to irradiation temperature and neutron flux/fluence due to the samples irradiation location.
- The modified surveillance programme causes inaccessibility of RPV wall in the container area and therefore in regions close to the active core for NDT.
- The evaluation of published surveillance results from WWER-1000 materials taking into account the estimated irradiation temperatures does result in strong uncertainties about the neutron embrittlement of WWER-1000 steel. From comparison with available data it is obvious that the specification in the Russian Code (A_F= 20 for welds, 23 for base material) cannot be considered conservative.

- Although the first reliable results will be available from the Temelín surveillance program, these results cannot eliminate the uncertainties about the WWER-1000 RPV steel embrittlement: the RPV specific surveillance program cannot provide a reliable statistics background for the prediction of the material degradation, since every set of samples withdrawn and evaluated provides for only one single data point to be added to the irradiation embrittlement versus time correlation.
- The embrittlement coefficients determined so far for Temelín specific materials are based on irradiation in test reactors with high lead factors. The existing dose rate effect might have adversely affected the determined embrittlement coefficients; the embrittlement might be higher in reality.
- The material properties data in the passports indicate that the initial critical temperature of brittleness T_{k0} can vary by tens of degrees from one weld metal charge to another. It has not been possible to check whether the temperature margin δT_M 10 [K] for the base material and 16 [K] for the weld metals) as defined within the VERLIFE methodology in order to cover the scatter of the mechanical property values has been taken into account.
- This fact and the uncertainties of the specified embrittlement coefficients need to be taken into consideration by using the safety factor ΔT as required by the IAEA Guidelines [IAEA 1997].[59]
- Weld no. 4 in ETE-1 was welded with two different electrode heat charges (Sv 12Ch2N2MAA, heat number 17084 and 170007) for both heat numbers surveillance samples were fabricated; the surveillance programme of ETE-1 is performed using the samples welded with the same electrode heat than weld no. 3 (Tk_0 = –50 [°C]). The other weld metal with T_{k0} = –30 [°C] will be irradiated within the surveillance programme of ETE-2. In view of the Austrian Experts' Team, this is a shortcoming because the results on irradiation embrittlement of the weld material with the highest T_{k0} of ETE-1 will not be available without significant delay.
- The fracture toughness curve formula used in the VERLIFE methodology can be considered conservative as compared with fracture toughness curves of other National Codes.

4.4 Issues of further interest, monitoring items concerning the surveillance programme – material embrittlement

- It would be of interest to compare the materials characteristics determined within the qualification tests, the extended acceptance tests and the lifetime evaluation programme cited during the Workshop [BRUMOVSKY 2004a] with the surveillance programme data in order to evaluate the scatter of materials characteristics.
- It is of importance in the future to monitor the results of the surveillance programme for both units. Special emphasis should be dedicated to the surveillance results of the weld no.4 samples (including the heat affected zone). The first results of the surveillance capsule removed in May 2004 will be available in 2005.
- It is recommended that the results of the surveillance samples irradiated in unit 2 (esp. specimens of weld no.4/unit-1 and weld no.4/unit2, including HAZ) should be monitored with special emphasis during the next years. At the same time it would be desirable to obtain information whether specimen of weld number 2 are included in the PTS considerations.
- The evaluation of experimental assessment of the neutron embrittlement of ETE materials using surveillance specimens should be included in a continuous monitoring in order to confirm the application of temperature margins (upper boundary of the radiation induced T_k shifts to be used in the RPV lifetime evaluation) as defined in the VERLIFE methodology.

[59] even in the draft of revision 2

5 NDT CONCEPT AND PROGRAM

Areas of Monitoring

No	VLI/VLI group description
3	**OPERATION & MAINTENANCE**
3.3	**IN SERVICE INSPECTION**
3.3.1	**ETE – ISI implementation and verification.**
1	Is all the relevant equipment subject to regular functional testing? What are the schedules for testing and/or inspections?
2	Are procedures in place to provide for adequate inspectability of PTS equipment relevant to safety? Where are these procedures documented? Is the application documented with the operator and approved by the licensing authority?
3.3.2	**ISI of PTS equipment relevant to safety**
1	Is the inspectability of all RPV sections and components adequate?
2	How and when were the RPVI related NDT programmes qualified?
3	Has a 100% RPV NDT test been performed with qualified methods in ETE-1 and ETE-2?
3.5	**MATERIAL EMBRITTLEMENT HISTORY VERIFICATION AND CONSEQUENCES**
3.5.2	**Tools for PTSA: Monitoring of the NDT results**
1	Which procedures for monitoring NDT results exist to provide for comparison options in time while evaluating the status of the RPV integrity and assessing the development of deficiencies with a high degree of confidence?
3.6	**EMBRITTLEMENT MANAGEMENT STRATEGIES**
3.6.3	**RPV integrity management: NDT program, qualification and application**
1	Is there a qualified NDT program for ETE-1 and ETE-2?
2	When has a qualified NDT program (if there is one) been applied for the first time in ETE-1 and ETE-2?
3	Was any instrumentation dedicated to PTS/RPVI installed in addition?
3.6.4	**RPV integrity management: NDT results monitoring**
1	What procedures are used for comparison of NDT monitoring results before implementation of a qualified NDT program and the qualified results?
2	Is there any possibility to evaluate the monitoring results of the non-qualified testing with respect to flaw development?

5.1 NDT Concept and Program (limited to considerations presented during the workshop – related to RPVI and PTS)

Remark: Non-destructive testing is a separate project (PN10), which will be treated also within in the roadmap.

Within project PN9 – RPVI and PTSA this issue will be treated in the present context, and consider only the presentation at the PN9 Workshop in Prague.

5.1.1 Description of the issue – fundamentals

Nuclear reactor pressure vessels are examined at regular intervals during in-service inspections (ISI) in order to detect service activated flaws and monitor their behaviour to make sure that no dangerous cracks remain undetected, also from fabrication and during pre-service inspections. Since detection methods are improved at all times it is possible to detect also flaws from manufacture during in-service inspection with modernised NDT techniques – it is also possible that flaws grow to a detectable size during service.

5.1.2 State-of-the-art requirements and regulations

The National Codes specify rules and regulations for the acceptability of detected indications.

<u>Russian Federation</u>

The regulatory requirements concerning the acceptance of flaws during ISI are defined in the “Methods for detection of permissible flaws in metal of equipment and pipelines during NPP operation”.

<u>United States</u>

The ASME Code, Section XI, acceptance standards, define the fracture mechanics methods to be used, the required safety factors may be applied on flaw size (IWB-3611) or on the stress intensity factor (IWB-3612).

<u>Germany</u>

According to KTA 3201.2 the acceptance of flaws is based on fracture mechanics verification considerations while checking against the fracture toughness curve with RT_{NDT} applicable for the RPV material at the time of the next inspection.

<u>France</u>

The French code distinguishes between fracture-mechanics assessment of harmful defects and defects that are acceptable without special analysis.

The fracture mechanics treatment is the one described for the PTS analysis (including the safety factors applied on K_{Ic}). The risk of excessive deformation and plastic instability must also be evaluated. A combined analysis of the risk of sudden defect extension and structural instability may also be performed using a “double criterion” method (RSEM B-5325).

5.1.3 Current plant status

Workshop presentations:

M. Brumovský, J. Žďárek: Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs; J. Sheibal: Statement on the in service inspection qualification process;

L. Horáček: Qualification of NDE respective to the PTS affected area of the RPV, V. Pištora: Comparison of IAEA, Russian and VERLIFE methodologies for PTS assessment

NDE programme within VERLIFE

With respect to the NDE programme of the RPV it was stated during the Workshop [BRUMOVSKY 2004a]:

"The inspections will be performed from both outside and inside of the RPV:

- *From inside by means of a special manipulator (SKIN) using ultrasonic, eddy current and visual inspection methods, and, if necessary, additional methods may be used;*
- *From outside special manipulator SK-187 will be used. The manipulator is being supplied as a part of the RPV delivery and it will enable to apply ultrasonic and visual inspections. Separate individual Quality Assurance Programmes were prepared and subsequently approved by the regulatory authorities for all types of RPV inspections.*
- *Non-destructive examination of RPVs during operation is a mandatory part of in-service inspections. In accordance with "Rules for Construction and Safe Operation of Equipment of NPPs, Experimental and Testing Nuclear Reactors and Devices", Gostekhnadzor, USSR (rem.: originally), 1973 (rev.1989)*
- *Interval of these inspections is 4 years. During this period, practically the whole vessel will be examined, both from the inner as well as from outer surfaces. The same extent and type, including method of examination has been performed as a part of the RPV "fingerprint" during pre-service inspections; the same manipulators and methods will be applied also during operational inspections."*

According to [BRUMOVSKY 2004a], the RPV integrity assurance is supported by several specific actions, such as an extensive NDE programme during manufacture and operation – in-service inspections including qualification.

Crack sizes/shapes

The postulated crack sizes and shapes according to the VERLIFE concept are contained in [PISTORA 2004a]: "semi elliptic underclad crack, provided that both integrity of cladding has been proved by NDE methods and mechanical properties of cladding are known".

"VERLIFE is conservative mainly with respect to the postulated crack size:

- It has higher depth of the crack,
- Due to semi-elliptical shape and $^{a}/_{c}$ ratio = 0,3, it has also higher maximum length.

As far as postulated crack size is concerned, the Russian methodology does not take into account performing and results of NDE of the weld or base material, neither it mentions qualification of the NDE method. The postulated crack size is 0,07.S. ….“For dimensions and fictive flaw shape reference, see the principal scheme on the next page.

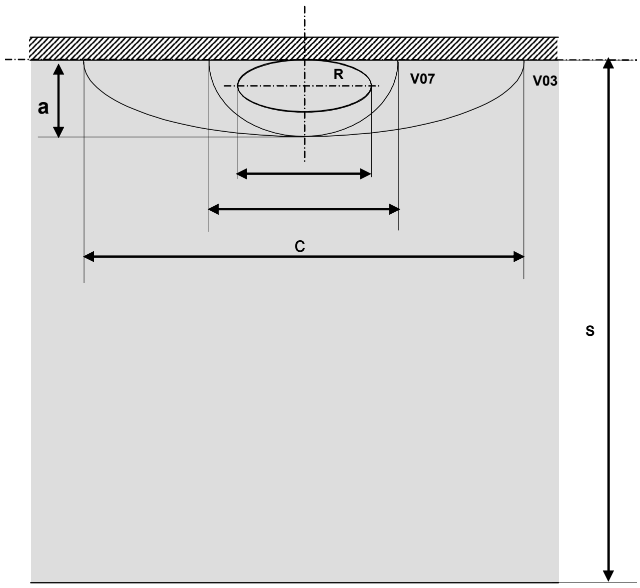

Figure 7: Dimensions and shapes simulated for flaws in the RPV wall as used in fracture-mechanics assessments

s............ Wall thickness R

a .... Crack depth V07

c............ Crack width V03

Table 5: Flaw sizes according to Russian code and the presumptive VERLIFE rule

PTS group	PTS scenario	T_k^a [°C]	approach (WPS/Tangent)	position (weld No.)	orientation	a/c	critical time [s]	critical point
Pilot study	2SLB	102.8	T	3	axial	0.3	2400	21
	SB32	86.3	W	4	axial	0.3	3000	20
	PSV1	61.2	W	4	circ.	0.7	2100	4
MSLB	SLB1A	126.9	W	3	axial	0.3	1650	21
	SLB1B	**108.6**	**W**	**3**	**axial**	**0.3**	**1980**	**21**
	SLB1C	111.2	W	3	axial	0.3	2950	21
PRISE	**3SGT**	**66.3**	**W**	**3**	**axial**	**0.3**	**3210**	**20**
	SGH1	89.4	W	4	circ.	0.7	2330	4
LOCA	**H850**	**102.7**	**W**	**4**	**circ.**	**0.3**	**1250**	**16**
PRZ SV	**PSV43**	**92.9**	**W**	**4**	**circ.**	**0.7**	**1730**	**4**
	PSV43B	**82.0**	**T**	**3**	**axial**	**0.3**	**2300**	**21**

NDE Qualification

The Czech Experts stated that qualification of the "ISI system in accordance with ENIQ (European Network for Inspection Qualification) guidelines and IAEA (International Atomic Energy Agency in Vienna) recommendations" [SHEIBAL 2004], [HORACEK 2004] is an on-going process:

<u>The status of the ISI qualification is according to the presentation:</u>

- Weld in reactor core area – successfully finished
- Nozzle homogeneous weld – will be finished 07/04
  (qualification dossier is not completed yet)
- Nozzle inner radius – will be finished in 08/04

Concerning the qualification criteria the following information was provided:

<u>Inner wall defects</u>

- Defects with a height (TWE) over 6,5 [mm] – required 100% detection
- Defects of height (TWE) in the interval from 3,0 ÷ 6,5 [mm] – required at least 80% detection
- Maximum allowed defect height (TWE);
  underestimation/overestimation error is ± 5 [mm]
- Maximum allowed defect height (TWE) RMS = 3 [mm]

<u>Surface breaking defects</u> (Near surface defects)

- Required detection of all defects of height (TWE) over 3 [mm]
- Maximum allowed defect height (TWE);
  underestimation/overestimation error is ± 5 [mm]
- Maximum allowed defect height (TWE) RMS = 3 [mm]

<u>Common criteria</u>

- Defects with a height below 3 [mm] are not of concern
- Defect length sizing maximum allowed; underestimation/overestimation is ± 10 [mm]
- Tolerance on the ligament sizing ± 4 [mm]
- Positioning along the peripheral (difference between real and measured position of the defect centre) is not allowed over ± 20 [mm]

<u>False calls</u>

- No false calls allowed for defects where 100% detection is required
- One false call allowed for the weld length of 3 [m] in case of defects where 80% detection is required

The test sample KB 190 for the qualification programme was described during the Workshop the specific and postulated defects (designed by NRI Řež, approved by SONS in accordance with SONS Guideline No. 01966) were discussed in detail:

<u>Specific defects</u>

- Underclad crack type defects
- Lack of fusion type defects

<u>Postulated defects</u>

- Fatigue cracks in the butt weld root area
- Cracks parallel to the WCL in the butt weld

The Qualification Dossier documentation was finished in January 2004.

"Within practical trials all the requirements on detection, positioning, sizing and characterization were met in compliance with the Qualification Criteria."

Results of Qualified Inspections in the plant:

- Qualified RPV butt welds ISI from inside and outside surfaces applied at Temelín NPP for the first time during the outage of Unit 1 in May 2004 (ISI vendor ŠKODA JS)
- Qualified inspection procedures, being in force since January 2004, include for the first time the examination of butt weld root area with TOFD[60] technique (supplementary to the pulse echo standard technique).
- Detail site feedback results from qualified inspection are expected be provided by ŠKODA at the beginning of June 2004.
- Detail site feedback experience also available from:
- The qualified inspection from outer surface performed on WWER-440 type RPV Unit 1 and 2 butt welds, base metal and cladding interface at Dukovany NPP in 2003.
- The vendor ŠKODA JS has supplied also the ISI equipment including sensors.
- Qualified inspection of WWER 440 type RPV performed successfully from inner surface by ŠKODA at Páks NPP in 2003.
- In all the above cases site feedback correspond to the experience obtained during the mechanised UT qualification.

5.1.4 Evaluation

The Western Code regulations require qualified NDE methods to be applied for the defect monitoring by ISI.

With respect to the "wide NDE programme during manufacture and operation", it is questionable whether the inspections have already been carried out or are only foreseen as future work. The actual status and schedule should be sorted out.

There appear to be problems related to NDT procedures qualification in relation to PTS requirements because – according to the time schedule – qualification procedures were carried out by the end of 2003 and reported in January 2004. Furthermore the same manipulators and methods are not very likely to be used, as those applied for the "fingerprint" pre-service inspection, and also during the qualification procedures mentioned before and the ISI in May 2004. That the UT equipment as applied is mainly suitable for TOFD based inspections adds to this assumption, but at the same time traditional Pulse-Echo UT-inspections (PET) can be performed.

For the RPV cylindrical wall the ISI NDT methods have been qualified successfully and be regarded to allow detecting all kinds of crack-like defects, which are of special concern for the PTS analysis, e.g. a crack close to the inner cladding with an $^{a}/_{c}$-ratio of 0,7 and different extensions into depth depending on the PTS assumptions. For the NDT the worst cases seem to be semi-elliptical cracks starting at the cladding interface and extending 8 [mm] deep into the ferritic wall.

The worst case condition is not linked to the detectability of those defects – this can be considered reliably proven by the qualification measures at the RPV wall test block –but it must be taken into account, that the clear proof for near cladding defects to be confined within the ferritic base material and not extending into the cladding cannot be derived based on the NDT techniques actually applied.

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[60] Time of Flight Diffraction technique an ultrasonic NDT method suitable to examine thick walled components

Although the qualification measures at the RPV wall test block demonstrated the basic potential of the applied UT methods to detect those defects, there is however the following remaining problem not yet solved: The test block does not contain the cladding condition at the welds and on it's vicinity, where one has to take into account a considerably higher noise level. This requires special countermeasures, e.g. additional Eddy Current testing in areas with an elevated number of UT indications. Since for the required Eddy Current method neither the qualification nor the inspection has been carried out yet, the safety assessment concerning – the absence of cracks relevant to PTS – has not been demonstrated at present by the results of qualification procedures.

According to the report [HORACEK 2004], the first experiences with the TOFD inspection are date back to 2003. Eddy current inspections are mentioned, but no qualification has been presented so far.

Mr. Brumovský confirmed during the Workshop discussions, that the past NDT measures and their qualification as well as their evaluation have to be adapted to the newly defined VERLIFE concept.

The treatment of listed critical defects for different PTS transients as analysed [PISTORA 2004b], is made apparently with the assumption the cladding to be intact – this then must be proven by qualified NDE. The qualification of the cladding and cladding ligament check as presented at the Workshop [HORACEK 2004], raises substantial questions in this respect. At the same time, it is not clear whether the licensing authority has accepted this as a sufficient confirmation for intact cladding qualification.

In view of all NDT applied at the near cladding region, it must be pointed out, that a discrimination of a detected indication according to elliptical and semi elliptical shape cannot be based on NDT results.

Different arguments are used for the definition of the critical defect sizes: At the RPV wall the PTS relevant crack sizes are considered, in other situations the standard thresholds of NDT rules are used, these are mostly derived from the performance limits of the given NDT technique with respect to the individual task. (e.g. for the inspection of the inner nozzle corner of the DN850 main coolant pipes connecting to the RPV the assumed thresholds of a 7 (10) [mm] deep crack are probably close to the capabilities of the foreseen UT method). It remains to be yet proven whether they are also in a conservative relationship to a PTS related critical crack size.

According to the presentation of the PTS analyses [PISTORA 2004b] the PTS-critical defect sizes at the cooling water inlet nozzle corners and connecting welds of the primary loop have not been investigated yet, but quantification of these critical defect sizes is needed for the NDT qualification procedures foreseen to be ready for use in July or August 2004.

The qualification criteria concerning surface breaking defects are not clear. What is meant by “surface breaking defect”? According to [PISTORA 2004b] for the PTS analysis the cracks considered do not open towards the wet inner surface, because the cladding is assumed intact. However, many of the defects of this type are in the test sample KB 190.

Due to a lack of more detailed information on probe positions, for the various types of defects, the results of the qualification cannot be judged; insufficient information was provided also on data acquisition and data presentation.

Due to a lack of more detailed information on probes, the results of the qualification cannot be judged; insufficient information was provided also on probe positions for the various types of defects, on data acquisition and data presentation. Therefore, the following evaluations are in part based on estimates on the techniques most likely applied by the Czech Experts.

The results of the qualification are somewhat difficult to be judged due to a lack of more detailed information on probes, data acquisition and data presentation. The ISI carried out in

May 2004 has presumably used the standard pulse echo probe arrangements (0°, 45° and 60° probes with the sensitivity thresholds already explained by the Czech colleagues during a previous meeting. This may correspond to ASME section XI procedures and may even be a more intensive inspection due to the inspection from the inner and outer side of the vessel. About the application of the 70° Transmitter/Receiver probes with inclined longitudinal waves (TRL 70°-Probes) – which are especially needed for the inner near cladding surface area – documented information is available since the October 2004 workshop only. It can now be assumed that they are operated with similar sensitivities as usually applied in western countries. Mr. Horáček argued in this way. The additional use of TOFD arrangements should help to size detected defects and serves also as a replacement of tandem techniques. A realistic comparison of the performance of the Tandem and the TOFD approach for internal defects perpendicular to the surface is not available. Certainly the application of TOFD gives a better proof for the absence of this kind of defects, but it can surely not replace corresponding inspections during fabrication and pre-service.

It is claimed that the detectability of defects perpendicular to the surface inside the wall at the cylindrical shell must be assured by angle beam probes and at the three circumferential welds additionally with a TOFD approach. Tandem probe arrangements especially suited for this purpose are not applied. It has been argued, that the redundancy of an inspection from in- and outside and in addition the TOFD approach are guaranteeing that no dangerous inner-wall cracks remain undetected.

Neither was there a special inspection conducted for those defects during the fabrication nor during pre-service (a fabrication X-ray inspection of the root areas with a Betatron cannot be regarded as a valuable replacement). In addition, it has to be recognized, that the recent TOFD application at the three circumferential welds of the RPV produced an unusual large number of indications (96 indications have been noticed by Škoda JS), which is normal for a TOFD technique but rather strange for an in-service inspection at an RPV. Given the fact that a normal TOFD application on very thick welds must end up with a fairly high amount of unclear indications (which are classified as to be out of interest only with the additional information from the standard pulse echo techniques), it remains unclear, whether the TOFD can in future be regarded as a realistic replacement of a tandem technique or not. The first experiences and data available indicate that this cannot be the case. The TOFD approach must therefore in future be restricted to analytical purposes in case of indications exceeding the analytical threshold and should not be used for detection tasks.

Škoda JS has apparently used the new Micropuls equipment as a front end of the UT measuring chain, which delivers the A-scan signals to a traditional gate based UT-apparatus developed and owned by Škoda (probably the one which has been used by Škoda already in the past). This certainly facilitates the application of a proven and well-established evaluation software.

A general statement from the last PN2 Workshop must again be repeated: A worst case defect for a fracture mechanics analysis is not equivalent to a defect representing worst case conditions for NDT detectability (see also the ENIQ Document EUR 1868 EN). This concerns all rectangular-shape defects in the test sample KB 190. These defects offer ideal and not worst-case diffraction conditions for the TOFD technique (see also [WÜSTENBERG 1997]). Therefore, all results of the qualification concerning the detectability and sizing of the corresponding defects with the TOFD technique are restricted to very special defect shapes and cannot be generalized.

Another more severe objection concerns the determination of the ligament, which is unfortunately not defined clearly in the contribution [HORACEK 2004], but describes probably the distance between the wet inner surface and the lowest tip of a defect. It has to be assumed that the possibilities of the TOFD approach are used for the sizing of the ligament.

Whereas for inner wall defects the general statement concerning rectangular-shape reflectors made above has to be regarded, for the near cladding defects another fact has to be considered. All defects starting at the interface and being extended into the base material can be detected by various techniques with a high detection probability (70° Transmitter/Receiver probes for L-waves from the ID and corner effect from the OD). If one tries to combine detection and sizing e.g. based on interactions using a crack tip diffraction (TOFD-Technique, SLIC-Probes from South-West-Research Institute), the detectability for underclad cracks is strongly reduced.

The report [HORACEK 2004] does not reveal the technique used for the detection of underclad cracks. The indicated detection limits for the near surface cracks can lead to the assumption, that one of the methods with optimum detection probability has been used, however, it must be assumed that due to the extensive use of the TOFD technique, detection and sizing have been combined using diffraction based techniques.

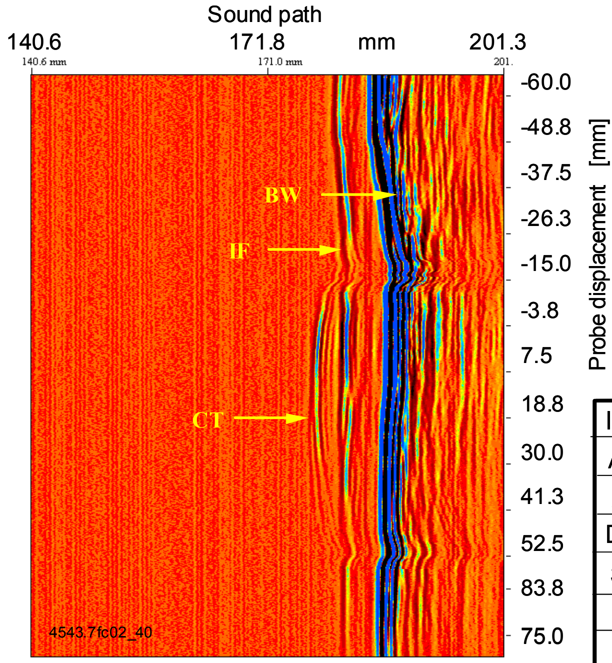
In addition the indication from the crack tip close to the interface is mostly buried in a noisy signal from the interface based on the difference of the acoustic impedance of the ferritic-bainitic base material and the austenitic cladding. This makes it very difficult and almost impossible to distinguish the lower tip of a near cladding crack from the interface in cases of realistic defects. The type of defects used for this situation within the test sample KB 190 are most probably not corresponding to such realistic conditions, they are not representing the typical worst case for the interface region.

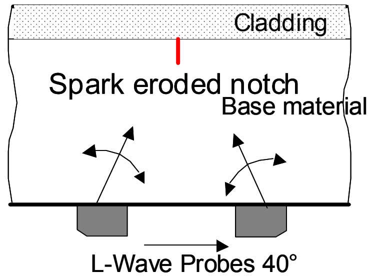
These objections are valid for an inspection from the outside as well as for the inside.

The following Figure 8 shows an example for a TOFD-inspection on a cladded reactor vessel wall with a typical underclad crack type test reflector. One may clearly recognize the upper crack tip and derive from it the depth extension in the base material but it is not possible to say anything about its extension towards the interface or into the cladding.

This limitation is the reason why it is necessary to apply a specialized low eddy current inspection from the inside in order to prove the integrity of the cladding. (see also [WÜSTENBERG 1996]). A corresponding qualification is not yet scheduled but obviously intended, as mentioned by Mr. Brumovský and Mr. Žďárek.

The qualification procedure for the RPV wall inspection using test sample KB 190 as presented suggest that the scanning and data presentation procedures are optimized during this trial. This assumption is also supported by the fact, that the nozzle inspection procedures are not yet finally optimized. During the visit at the NRI in Řež, Mr. Horáček explained, that the TOFD application qualification had to be repeated because the Škoda operators were not enough experienced with this approach. Although in essence this cannot be criticised, it indicates that the comparability with a "fingerprint" inspection during pre-service may be very limited.





Notch dimensions:
Depth x Length (6 x 24 mm²)

Inspected object	RPV at the MPA
Angle of incidence	40° (LW)
Amplification	70 dB
Distance of the probes	240 mm
Signal/Noise-Ratio	21 dB
Frequency	3,7 MHz
Defect-Type	Undercladding Crack
Wall-Thickness	145 mm

CT = Crack Tip
BW = Back Wall
IF = Inter-Face

Measured extension (Distance Crack Tip - Interface): 5 mm

TOFD at a cladded testblock

BAM

Figure 8: TOFD applied to a cladded test block

In order to clarify some of the critical items (e.g. the value of the statements concerning the ligament), it has been proposed to use the special NDT workshop in Řež foreseen for October 2004.

5.2 Conclusions concerning the NDT Concept and Program

- The ISI with ultrasonic NDT methods for the RPV cylindrical wall has successfully been qualified and can as such be regarded to basically enable detection of all kinds of crack-like defects, which are of special concern for the PTS events and their analyses, e.g. a crack close to the cladding interface to the base material layer with an aspect-ratio $^a/_c$ of e.g. 0,3 and different extensions in depth depending on the PTS assumptions. A semi-elliptical crack seems to be the worst case for NDT, which starts at the cladding interface and extends 8 [mm] deep into the ferritic wall. Although qualification using the RPV wall test block demonstrated the basic potential of the applied UT methods to allow detection of those defects, there remain some problems not yet finally solved.
- The test block does not contain the cladding condition at the welds and on its vicinity, where one has to take into account a considerably higher noise level and therefore a higher false call rate and this is mentioned by the qualification report. This requires special countermeasures, e.g. additional Eddy Current Testing (ECT) in areas with an elevated number of UT indications. This is particularly needed, because the VERLIFE concept requires a sound cladding, especially at locations of near cladding cracks in the ferritic wall.

The remaining ligament between the lower crack tip and the wet inner surface can only be proven with appropriately qualified ECT methods. Since neither the qualification of, nor the inspection with the ECT method as required has been carried out yet, the safety argumentation concerning the absence of PTS relevant cracks is not closed at the present time.

- Two other ISI areas bearing specific PTS concerns are the inner corner of the inlet nozzles and the welds connecting the primary loop to the RPV. For both areas, qualifications have been announced but have not been finished yet and presented. Of special interest are the PTS relevant crack sizes within the nozzle corner and the connecting weld, in order to judge the difficulties the NDT techniques will have to guarantee sufficient detectability and a reasonable false call rate.
- In view of the remaining NDT activities not yet finished, but needed to prove the absence of all kind of PTS relevant cracks, one must conclude, that the NDT inspections carried out until today cover only in part all the ISIs required. According to the information given at the PN9 Workshop the completeness of the ISI concerning the PTS analysis is in preparation, with several qualification activities ongoing, but will certainly not be reached before the foreseeable next RPV ISI.

5.3 Issues of further interest, monitoring items about the NDT Concept and Program

This issue of NDT will be treated in detail in project PN10: Integrity of Primary Loop Components – Non Destructive Testing (NDT) [Item No. 4] [APPEDIX C].

6 MITIGATION MEASURES

Areas of Monitoring

No	VLI/VLI group description
3	**OPERATION & MAINTENANCE**
3.1	**OPERATION AND Performance of equipment**
3.1.2	**Status of symptom based emergency operating procedures (EOPs) related to PTS/RPVI, and of Severe Accident Management Guidelines (SAMGs)**
1	What is the current implementation status of those elements related to PTS/RPVI, which ones are to be used together with EOPs and SAMGs? Does the training currently provided treat this implementation as an issue?
3.6	**EMBRITTLEMENT MANAGEMENT STRATEGIES**
3.6.1	**Feedback from PTS experience into the related training**
1	Is all the important information needed for PTS/RPVI management available in the MCR and at the on-site emergency centre (TSC) as well?
2	Is guidance being provided for identification and optimisation of strategies, actions, and plant features used in PTS conditions? Does the provided guidance include the assessment of equipment and instrumentation? Was any operational aid developed for PTS/RPVI purposes (to compensate for insufficient information)?
3.6.2	**RPV mitigative design: Core design**
1	Which requirements were specified for the change of the core configuration with respect to the original design? Has the core design been changed for RPV-wall irradiation minimisation?
2	Has a low leakage core been implemented? If not yet implemented, what is the anticipated schedule for accomplishing this? If It is not planned to configure a low leakage core for Temelín, what is the assessment basis for concluding that this is not necessary for the planned operating lifetime of the reactors?
3	Which kind of fuel loading pattern and refuelling strategy has been adopted for Temelín? Does this strategy affect thermal power rating? Which changes are envisaged in order to compensate for eventual power reduction?
4	Have additional absorbers been included? Are those absorbers integrated into the fuel elements or are they part of the core barrel?
5	What is the current position of the operator on an eventual need for corrective action regarding the materials properties in case they deteriorate?
3.6.5	**RPV integrity management: PTS mitigative design/operational provisions**
1	Have the high level AM strategies related to RPVI ant PTS mitigation been implemented as readily usable procedures/guidelines (SAMGs) and transition options from EOPs?
2	What analyses have been performed of RPV failure (due to PTS) in terms of the potential for RPV missiles to be generated? What RPV thrust forces have been calculated for a spectrum of RPV failure defect sizes? How do these thrust forces compare with the force required to cause failure of the RPV supports and the RCS piping?
3	What analyses have been performed of the structural integrity of the RCS piping in case of RPV failure due to PTS?
4	What is the design pressure of the reactor cavity? What range of RPV effective leakage l sizes can be accommodated before the design pressure of the reactor cavity is exceeded?
5	What analyses have been performed of the structural integrity of the reactor cavity area in case of RPV failure due to PTS? Have these analyses considered the potential for an RPV lower head missile generated as a result of a circumferential failure of the RPV due to PTS? Can the reactor cavity floor accommodate such a missile without structural failure?

No	VLI/VLI group description
3	**OPERATION & MAINTENANCE**
6	What are the AM strategies applicable upon transition from RPV failure due to PTS into the SAMGs? Are the operators directed to follow SAG-3 ("Inject into the RCS")? If so, what are the limits of the effectiveness of available RCS, injection sources in terms of RPV break location and size, and are these limits indicated in SAG-3?
7	What analyses have been performed of the severe accident progression, which would ensue after RPV failure due to PTS? What is the effect on accident progression of the ingress of air into the RPV through the RPV effective leakage sizes?
8	Do AM procedures include any measures related to possible short-term impacts of RPV failure?
9	Which precautions are taken in relation to reduction of radioactive releases to the environment that could be considered specific to the nature of PTS induced loss of RPVI?
3.6.6	**EOPs and SAMGs: Required Control Room procedures and guidelines for RPVI/PCSI**
1	Following entry into EOP E-0 ("Reactor Trip or ESF Actuation"), into what EOPs would the operators transition, and would they then reach either FR-P.1 ("Response to Imminent Pressurized Thermal Shock Conditions") or FR-P.2 ("Response to Anticipated Pressurized Thermal Shock Conditions")?
2	What operator interventions are identified in EOP FR-P.1, "Response to Imminent Pressurized Thermal Shock Conditions"? What cues exist to direct the operators to perform these interventions in each case? What cautionary statements exist in this EOP to aid in avoiding possible interventions, which could make worse the evolution of the transient (in terms of PTS)?
3	What operator interventions are identified in EOP FR-P.2, "Response to Anticipated Pressurized Thermal Shock Conditions"? What cues exist to direct the operators to perform these interventions in each case? What cautionary statements exist in this EOP to aid in avoiding possible interventions, which could make worse the evolution of the transient (in terms of PTS)?
4	Are there any transitions from EOP FR-P.1 or FR-P.2 to the SAMGs? If there were none, would it not in fact require RPV rupture or leak followed by core heat-up above 650 °C before transition to the SAMGs would occur?

6.1 Core design – neutron fluence measurements

6.1.1 Description of the issue – fundamentals

Fluence estimates calculated at the RPV wall are very sensitive to the calculation procedures. Because of high neutron fluence attenuation between the core and RPV the calculated fluence is also strongly sensitive to the physical model of the core and RPV internals as well as to the mathematical model for the neutron transport calculations. The accurate determination of the RPV fluence is difficult and comparisons of measured and calculated data show a varying degree of agreement for different WWER designs and different core loading schemes.

6.1.2 Current plant status

Workshop presentations: M.Mečíř: RPV Fluence Minimization

The presentation covered only very general information on the implications of the reactor vessel fluence:

- Complex solution
- Safety implications
- Economic implications
- Fuel cycle cost
- Core design methodology

With respect to the treatment of RPV fluence a “Conservative approach” is adopted, including:

- Fluence calculation
- Power distribution
- OUT – IN strategy
- Core neutron leakage
- Geometry consideration
- Weighting factors

The concluding statement was: “Operation well below fluence calculation input” – presumably input for embrittlement prediction.

6.1.3 Evaluation

In ETE, fact is that Westinghouse has implemented a new core concept replacing the original concept of the Russian designer. It is not known whether this concept has been validated since it is sort of a prototype arrangement. Until construction of ETE there has been no essential core modification made to the original Russian design.

No information was made available on whether the core design has been or will be modified for neutron fluence reduction at the RPV wall. Because the Czech Experts suppose low neutron embrittlement, there is obviously no actual consideration of a core modification. The concluding remark that the operational fluence values will be below the calculated values is supporting this presumption.

6.2 Radiation embrittlement mitigation

6.2.1 Description of the issue – fundamentals

The definite program and the implementation of radiation embrittlement mitigation, and this way of avoiding PTS consequences from becoming safety critical and/or lifetime decisive for the plant’s units at NPP Temelín is an important issue in the frame of PTS mitigation measures.

The options to control radiation embrittlement or the adverse consequences to the RPVI are best described in the following Figure 9.

In this context, the efforts must be mentioned, suitable to keep high neutron fission efficiency and at the same time turn down the fluence effects to the RPV wall.

Not only the loading patterns but also collateral measures in adjusting the enrichment and adding neutron absorbing materials and burnable poison in the appropriate core locations are used. The loading pattern refers to the procedures adopted when refuelling the reactor for continuation of operation. It depends on the period of refuelling, the usage – burn up – of the fuel in the past, the type of new fuel inserted etc. Some calculations for reconfiguration of the core are needed for efficiency and neutron leakage and at the same time embrittlement considerations.

Evasion of PTS occurring is also applied after careful analyses have identified all possible causes. In most cases, the coolant conveyed for the various purposes to the PCL is adjusted in temperature to the actual state of the RPV.

The mechanical properties of the RPV wall material are also a candidate for improvement. However, most of the treatment options are restricted in their application first because treatment in on site and the boundary conditions required are difficult to keep because of accessibility and the large size of the vessel. In this respect, also the annealing must be considered.

The understanding of deficiencies or flaws as potential crack initiators has helped in developing suitable instruments to follow the development of the RPV with service time. Thorough analyses with a hoist of NDT techniques addressing different suspected deterioration effects have been developed and are in use. The well understood embrittlement process together with destructive testing of specimen stored in the RPV close to the core barrel are the appropriate means for determining the material properties during life-time of the RPV.

Load management also plays an important role in PTS consequences avoidance and for RPVI. These are operation temperature change gradients' limitations in between other. This option is evidently the choice of a number of NPPs including ETE. The required steps for introduction are related to fuel vendor low-leakage management schemes, like Babcock & Wilcox originating Lumped Burnable Poison (LBP) IN-OUT-IN or IN-IN-OUT featuring 30 ÷ 40% or locally 50% fluence reduction, Combustion Engineering SAV-FUEL is an IN-IN-OUT scheme with 20% reduction, Exxon's Low Radial Leakage is a mixed strategy with locally 50% fluence reduction and the Westinghouse Low Leakage Loading Pattern (L3P or LLLP) an IN-OUT-IN scheme with 10 ÷ 50% fluence reduction.

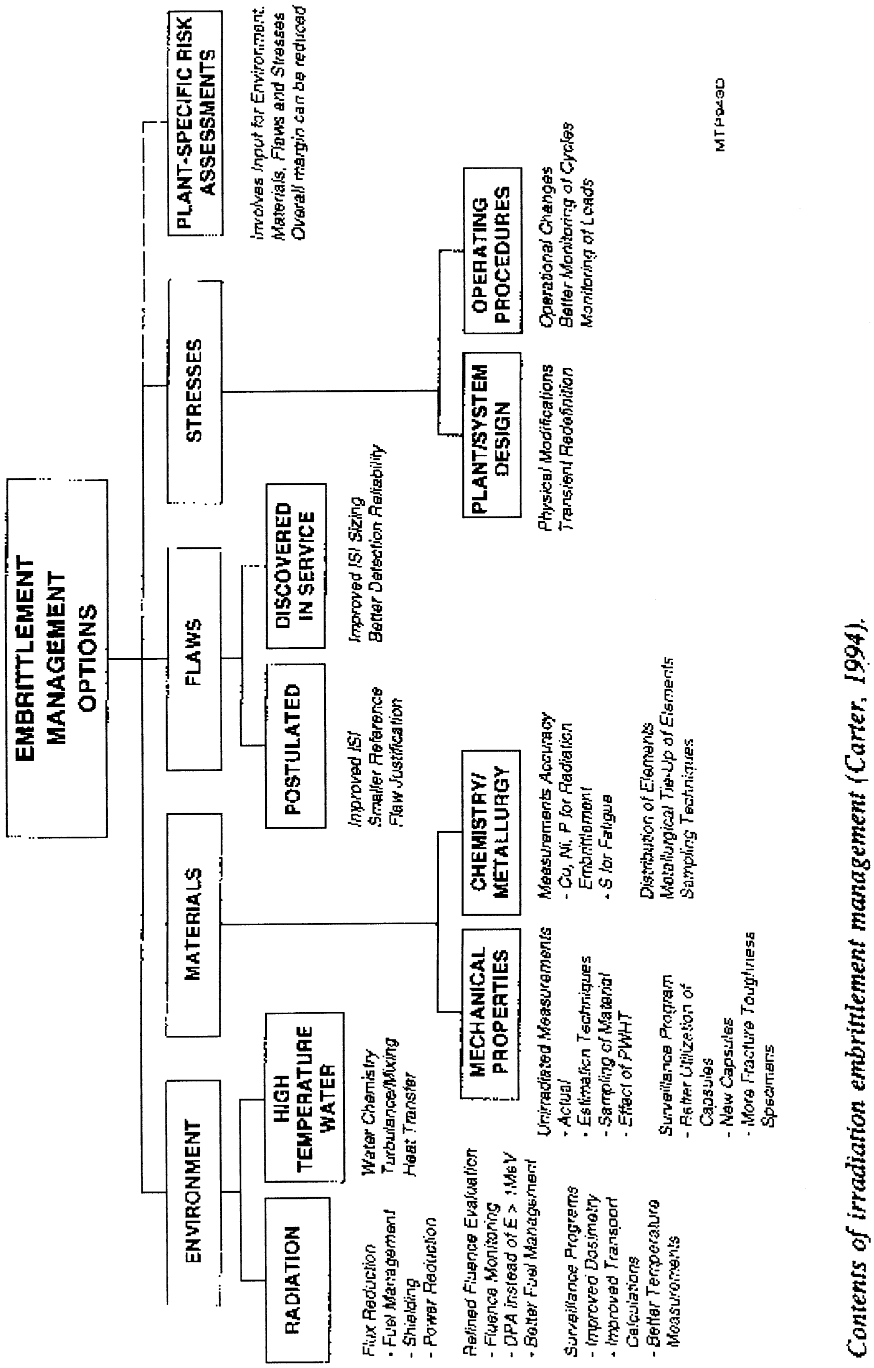


Figure 9: Embrittlement Management Options [CARTER 1994]

Some of the low leakage schemes are predominantly intended to improve fuel cycle economics, since fluence reduction factors up to 3÷5 can be achieved without a power reduction need. Some schemes make use also of 4-cycle fuel at selected locations close to the welds. The peripheral dummy assemblies are good for a reduction of up to 95% Figure 9.

Table 6: Exemplary Embrittlement reduction by loading pattern selection [AMES1 1994]

LOADING PATTERN	MAXIMUM FLUENCE RATE REDUCTION (%)
OUT-IN-IN (REFERENCE)	0
IN-OUT-IN	30%
IN-IN-OUT	40%
IN-IN-OUT: Max. local	50%
OUT-IN-IN: 4-cycle fuel at selected locations	60% locally
IN-IN-OUT: 4-cycle fuel at selected locations	70% locally
IN-IN-OUT: 4-cycle fuel at welds and control rods in the assemblies at the welds	90% at weld
Dummy peripheral assemblies	90-95%

6.2.2 Current plant status

Workshop presentation: Mečíř V.: Reactor vessel fluence reduction

It can be concluded that fluence reduction will be managed applying an OUT-IN strategy using a sophisticated core reshuffling technique, that is based on power distribution corrected fluence calculations and takes into consideration core geometry. The exemplary weighting factors as presented for the 45° core sector I/II outer fuel elements indicate the resulting power output shape and suggest a rather efficient embrittlement management trending application. The azimuthal as well as the radial power distribution rating characteristics for individual fuel element positions and elevations indicate also efficiency of the embrittlement management, where the outer fuel assemblies are rated on an approximate average power output of 80%, the radial rating as presented is between bounds of 80 ÷ 120%, which strongly indicates that economic considerations were also met by this management application.

The concluding statement “operation will take place well below fluence calculation input” suggests that embrittlement is managed properly, once the stipulated RPV fluence reduction management policy is enacted. At the Bilateral Meeting (PM5) a brief statement by Mr. Holan indicated, that an effort be made to introduce a Low-Leakage-Core starting from one of the next upcoming refuelling operations, with the equilibrium core established.

6.2.3 Evaluation

The supporting information does not allow for any further qualification of the approach than the one given below, since it provides a very limited insight into the real background of the presented expectations. The verbal communication was also provided in rather brief statements, suggesting the development has not yet reached the state of a conclusive decision that can be communicated. The material (diagrams and descriptions) are drawn from background information and reduced to basic essentials. The Experts’ Team was prepared to see a presentation of conclusive approach, indicating also, how evidently intended embrittlement mitigation effects could be achieved.

RPV fluence reduction management is envisaged by application of an option based on the OUT-IN strategy. When compared with the efficiency provided by the reshuffling schemes, the ETE scheme seems to fall short in effectivity. A reason for selection of such a policy was not provided. The implications of the change in core physics are well understood and a balanced approach evidently has been found, whilst associating economic demands with the recognised needs of plant ageing management, it is a very advanced approach using power distribution fluence control. The effect on fluence however was not clarified since deduction from the mere fuel bundle power output reduction rate is not an appropriate way of presenting transparent results for fluence reduction.

The very brief comment on the introduction of a Low-Leakage-Core setting as soon as the equilibrium core will be established, therefore starting after one of the next refuelling outages, is more than a positive sign for the handling of premature embrittlement and the associate PTS problems.

6.3 EOPs

Remark: This chapter also contains monitoring of the related training procedures (limited to considerations presented within PN9)

6.3.1 Description of the issue – fundamentals

Emergency Operating Procedures (EOPs) are supposed to identify and allow for control potential PTS conditions and to support the operator to safely shutdown the reactor without catastrophic consequences.

The internationally accepted approaches regarding RPVI/PTS are related to the general concepts realized with the introduction of EOPs, in most cases symptom oriented instructions for emergency management. EOPs allow for the handling of anticipated PTS events and for inadvertent PTS situations as well.

Upon occurrence of an initiating event or system failure potentially giving rise to pressurized thermal shock conditions and resulting in a reactor scram, the Emergency Operating Procedures (EOPs) would be entered. Entry into the EOPs occurs with every unplanned scram. The Severe Accident Management Guidelines (SAMGs) would not be entered until much later, and then only if the initiating event progressed to conditions where inadequate core cooling was threatened.

For the more general RPVI impairment cases, in particular those arising in severe accident conditions, appropriate SAMGs must be implemented, which take also into account Thermal Shock resulting from rapid cooldown actions or events.

6.3.2 State-of-the-art requirements and regulations

The state-of-the-art in procedural aspects of pressurized thermal shock (PTS) is to have symptom-based Emergency Operating Procedures (EOPs) in place to identify and control potential PTS conditions and bring the plant to safe shutdown without reactor coolant system pressure boundary failure and without the occurrence of inadequate core cooling. The goal of the EOPs is to avoid core damage, corresponding to the third level of defence-in-depth. Should conditions giving rise to core damage nonetheless occur, Severe Accident Management Guidelines (SAMGs) are required to be available to limit core damage and mitigate the consequences of such core damage. The goal of the SAMGs is to avoid large releases of radioactivity to the environment, corresponding to the fourth level of defence-in-depth. In the fifth level of defence-in-depth, offsite emergency plans are required to be prepared in order to limit exposures to the public and damage to the environment should a large release of radioactivity nonetheless occur.

Such recommendations have been followed in Western countries' NPPs on a mostly voluntary basis. Ever since the introduction of the EOPs/SAMGs, one of the issues to be observed was considered the PTS events and RPVI. In order to avoid loss of integrity of the RPV the recommendations of IAEA pertain also to include considerations about PTS. [IAEA 1997]

6.3.3 Current plant status

The EOPs at Temelín were developed under contract to ČEZ by Westinghouse, and were based on the Westinghouse guidelines in use at pressurized water reactor NPPs throughout the world. The EOPs are symptom-based, and they were implemented at Temelín in 1998. Recovery actions from PTS that are included in the EOPs include:

a. Stopping the cooldown of the reactor coolant system,
b. Checking to see whether a pressuriser safety valve or PORV should be closed,
c. Terminating safety injection if necessary criteria are satisfied,
d. Depressurizing the reactor coolant system to minimize pressure stress,
e. Establishing normal stable operating conditions in the reactor coolant system, and
f. Performing a thermal "soak" prior to further cooldown at restricted rate.

Eventually, the reactor cooldown and depressurization would proceed to cold shutdown conditions according to the Optimal Recovery Procedures in the EOPs.

Pressurized Thermal Shock (PTS) Events

Events Leading to PTS Conditions can be the outcome of two separate types of events

Extensive and rapid temperature drop in the RCS causing huge thermal stresses in RPV wall, the Pressurized Thermal Shock (PTS) Events and extensive and rapid pressure increase at low RCS temperature identified as Cold Overpressure (CO) Events.

Overview

Operation related PTS activities are set up according to the analyses of events leading to PTS conditions. The procedures are implemented and during EOPs training, using the Temelín Simulator the operational crews are taught how to interact properly in the selected PTS scenarios. With its capability adapted by the end of 2005 to a Full Scope Simulators also the PTS specific set of procedures will be available for training in its updated form.

Out of a total set of 40 Temelín EOPs and the interrelated 6 CSF trees 4 of the Optimal Recovery Procedures (ORP) apply in the various PTS emergencies.

Both types of events have been considered, the Pressurized Thermal Shock (PTS) Events and the Cold Over-perssurisation (CO) Events and for both management procedures have been developed. Event identification is based on standard measurements CL temperature, RCS pressure, RCS cool-down rate, and based on this the response actions are: stop cool-down and decrease RCS pressure in the PTS and immediate RCS pressure reduction in the CO cases.

The EOPs are generally oriented towards maintaining the 6 Critical Safety Functions; in case of PTS events these are predominantly sub-criticality (F-0.1), Core Cooling (F-0.2), Integrity (F-0.4), the Temelín function restoration procedures:

- FR-P.1 Response to Imminent Pressurized Thermal Shock Conditions and
- FR-P.2 Response to Anticipated Pressurized Thermal Shock Conditions

Both procedures have been designed for this function restoration purpose.

The recovery actions used in the PTS and CO cases are the following:

- PTS and CO Recovery Actions
- Stop RCS cooldown
- The recovery procedures include instructions allowing for:
- Identification of any possible source of excessive cooldown
- Termination or limitation of cooldown
- Checks if PRZR SV or PORV Should be Closed
- Identification of any open PRZR SV or PORV
- Actions derived from detailed PTS analyses based on VERLIFE methodology[61]:
- Closure of any open PRZR SV or PORV if should be closed
- Predefine RNO actions for late PRZR SV closure
- Terminate SI if Criteria Satisfied
- Control SI injection flow that significantly contributes to RPV temperature decrease and to high pressure conditions
- Determination if SI injection can be terminated (subcooling based on core exit temperature and RVLIS indication)
- Stop of HHSI to remove unfavourable PTS effect.

These operation and mitigative management actions are included into the training programs implemented at the Temelín NPP. This program was integrated later into the general EOPs training described here in brief, followed by a more detailed description of PTS related topics.

Training in general for EOPs

The EOPs Training involves the Control Room personnel as well as personnel involved in the emergency response structures up to the plant management. It provides the Trainees with an insight into EOPs philosophy, scope of coverage, rules of usage (user's guide) initial conditions and strategies used for recovery actions. It is a combined training with class sessions and training on the full scope simulator.

For the different operational regimes the following scenarios are within the training scope:

37 scenarios for normal operation conditions
58 scenarios for abnormal operation conditions
16 scenarios for emergency operation conditions

Training Program specifically for PTS:

The training program modules related to the PTS mitigation include the following:

- Scenarios/tasks included in training program, which could lead to PTS Steam line and FW line breaks
- All types of LOCAs
- All types of primary-to-secondary breaks

[61] The VERLIFE methodology extension into PTS analyses was mentioned in [Sýkora 2004]

- Inadvertent actuation of ECCS
- Inadvertent accumulator injection
- Malfunction of normal charging system
- Major scenarios/tasks which require CR staff interventions against PTS conditions (FR-P.1, FR-P.2 procedures)
- SLB outside containment on all steam lines (room A820) upstream of MSIV
- LOCA (equivalent diameter app. 45 mm) on CL compensated by HHSI pumps
- LOCA (equivalent diameter app. 10 mm) compensated by T_k charging pumps and consequently break enlargement up to 60 mm
- PRZR Safety valve opening.

Temelín Full Scope Simulator

The Temelín Full Scope Simulator (supplier: ORGŘEŽ SC Brno, CZ) is capable of simulating events with PTS implications. The capability[62] was validated in 2000, an upgrade is planned for the beginning of 2005 to Full scope simulator capability, then enabling to simulate:

- Normal operation conditions – unit start up and shut down (from cold shutdown to full power conditions)
- Abnormal operation conditions – transients with power reduction and systems malfunction
- Emergency operation conditions – accidents starting with reactor trip and/or ESF actuation[63]

[62] Severe accidents are out of scope of the existing model.

[63] Usage of two-phase flow model in all main pipelines.

Table 7: Full Scope Simulator relevant PTS related transient including EOPs application

Scenario	Event Probability:	Expected use of Procedures*	PTS conditions:	Capability Temelín Full Scope Simulator
SLB outside containment on all steam lines (room A820) upstream of MSIV	$2{,}3.10^{-2}$ [1/a] (inadvertent opening of SGSV) 10^{-4} [1/a] (1 SLB) 10^{-6} [1/a] (all SLBs)**	E-0 E-2 ECA-2.1 FR-P.1	- RCS pressure 12÷13 [MPa] (above HHSI pump shutoff head), - RCS cold leg temperature 100÷110 [°C] because of high RCS sub-cooling margin HHSI pumps are stopped - RCS is depressurized based on EOPs actions- PTS conditions are eliminated	In scope
LOCA (ED app. 45 [mm] on cold leg compensated by HHSI pumps	$4{,}5.10^{-4}$ [1/a]	E-0 E-1 ES-1.2 FR-P.1	- RCS pressure 6÷8 [MPa] (below HHSI pump shutoff head) - RCS cold leg temperature 100÷110 [°C] (loop with LOCA) - non-symmetric RCS cooldown caused by HHSI injection during HHSI flow reduction (PTS conditions), - HHSI flow is terminated because of sufficient RCS subcooling margin and RCS is depressurized based on EOPs actions - PTS conditions are eliminated	In scope
LOCA (equivalent diameter app. 10 [mm]) compensated by TK charging pumps and consequently break enlargement up to 60 [mm]	$7{,}5.10^{-2}$ [1/a] (ED 10 [mm]) $4{,}5.10^{-4}$ [1/a] (ED 60 [mm])	E-0 ES-0.1 ES-1.2 FR-P.1	- RCS pressure 12÷13 [MPa] (above HHSI pump shutoff head) - RCS cold leg temperature 260÷280 [°C]. - After break enlargement RCS pressure decreases below HHSI shutoff head - RCS temperature decreases - non-symmetric RCS cooldown caused by HHSI injection after manual start of two HHSI pumps (PTS conditions) - HHSI flow is terminated because of sufficient RCS sub-cooling margin - RCS is depressurized based on EOPs actions - PTS conditions are eliminated	In scope
PRZR Safety valve opening	$4{,}5.10^{-4}$ [1/a]	E-0 E-1 ES-1.2	- RCS pressure decreases below HHSI pump shutoff head - RCS cold leg temperature decreases - PRZR is fully filled by water - non-symmetric RCS cooldown does not exist - PTS conditions not satisfied - PRZR safety valve opening with late re-closure	In upgrade
New scenarios			Tasks will be prepared	

** Procedures acronym as used in the EOPs context*

*** Estimated value, not included into the Temelín PSA report)*

Having prepared and introduced all these measures for RPVI/PTS management the operator of the Temelín NPP draws the following conclusions:

The Temelín EOPs cover all possible types of events leading to PTS conditions. The Temelín full scope simulator capabilities are sufficient to train personnel (control room and TSC personnel) for events leading to PTS conditions.

All involved personnel obtain sufficient training since the Temelín EOPs Training Program covers all events possibly leading to PTS conditions.

6.3.4 Evaluation

ČEZ has adopted a standard and well-recognized procedural approach to PTS events in implementing Westinghouse EOPs and SAMGs. The EOPs were implemented in 1998, and the SAMGs are scheduled be implemented by the end of 2004.

RPVI impairment cases under severe accident conditions and the related SAMGs and SAM related possible countermeasures as implemented have been discussed to some extent in project PN7.

The presentation on EOPs and training has provided a broad overview on implementation management and feedback from dedicated analyses. It at the same time showed a close interrelation between different approaches to tackle the PTS avoidance and mitigation problem, first by generic approaches used for PWRs combined into symptom oriented EOPs of e.g. Westinghouse origin, secondly with the appropriate adaptations resulting from more in depth knowledge of vintage and plant specific behaviour that results from in depth analyses conducted and qualified.

Preparations made as well as the training programs developed and implemented have been much in line with the first year of operation and the implementation of the EOPs concept adopted.

6.4 Conclusions regarding Mitigation Measures

With regard to Core Design and Radiation Embrittlement Mitigation:

- The OUT-IN strategy is a well-known early means of embrittlement mitigation; the ETE specific information contained in the presentation did not give a clue to the question whether introduction is made for irradiation embrittlement mitigation, or just as a side effect of power output optimization. The PTS relevant effects of the RPV fluence reduction management can be derived from the power distribution sketches only. All other information was not available and it was not possible to determine to what extent the restricted information would have had to be considered really proprietary.
- The concluding statement “operation will take place well below fluence calculation input” does not per se endorse that embrittlement is managed properly. The RPV fluence reduction management policy is one element to be enacted along with plant operation.
- It is not known whether the concept of a Westinghouse core design within a Russian type reactor has been validated. Especially the fluence estimates for the modified core have not been discussed at the Workshop. The core design has not yet been modified aiming to a fluence minimization at the reactor pressure vessel wall in order to reduce the neutron embrittlement of the steel. This improvement will be implemented at one of the upcoming refuelling outages of the core. To date the intended changes have not been presented for monitoring.

With regard to EOPs and SAMGs transition:

- Extensive feedback from plant analyses was used to more appropriately adapt the EOPs outline and elements to an up-to-date emergency management tool. It can be understood from the overview presentation, that the concept is suitable for proper adaptation. This work is evidently a successfully ongoing process.
- The EOPs training and implementation activities are comprehensive and compare well with activities in other NPPs in Europe. In some instances, thoroughness was eventually given precedence before timeliness when implementing EOPs training opportunities.
- The EOPs as well as the SAMGs and appropriate precautions are set up according to State-of-the-art technology albeit the equipment to be used is qualified or has been qualified for the intended use in the respective operational regime.

6.5 Issues of further interest, monitoring items regarding Mitigation Measures

With regard to Core Design and Radiation Embrittlement Mitigation:

- The Temelín RPV embrittlement mitigation is of utmost importance for RPVI as long as the mechanisms and embrittlement progression are not known well enough to step down precautions. Fuel reloading patterns as well as changes in fuel composition and enrichment influence the neutron embrittlement deteriorating effects on the RPV; changes are envisaged after one of the next campaigns. The information provided is coarse requiring additional explanations, which would serve to answer the essential parts of the questions raised.
- It is recommended to monitor in the future whether a core modification aimed to reduce the neutron fluence at the RPV wall will be considered and implemented.

With regard to EOPs and SAMGs transition:

- The Temelín EOPs appear to rely on detection of a cooldown rate in excess of 60 [K/h] as an indication of PTS conditions. It is unclear whether this excess cooldown rate is expected to be noticed by the operators, or is detected by the plant I&C and indicated to the operators by appropriate alarms. In addition, it is not clear why plant status monitoring could not be performed using the I&C in order to provide an indication of potential impending PTS conditions before an excess rate of cooldown actually "arrives" at the reactor vessel.
- Given that the adopted procedural framework for PTS is based on the well recognized and well-accepted Westinghouse EOP/SAMG approaches, the matters identified above are recommended be pursued within the pertinent framework of the pertinent bilateral Agreement between Austria and the Czech Republic.

7 QA PROCEDURES

Areas of Monitoring

	VLI/VLI group description
3	**OPERATION & MAINTENANCE**
3.1	**OPERATION AND PERFORMANCE OF EQUIPMENT**
3.1.1	**ETE-PTSA implementation for operation**
1	Have PTS avoidance and mitigation measures exercises been conducted? If not, what arrangements are planned?
2	Are the exercises properly designed in order to make certain proper the PTS operation? Which scenarios are selected for the exercises?
3	Is the documentation of the exercises comprehensive (covering the preparation, conduct, results, insights, conclusions and feed-back)?
4	Did the selection of accident scenarios for the validation exercises allow for testing all relevant parts of the PTSA findings and the roles of different users?
5	What administrative arrangements have been introduced at the plant to control the process of PTSA implementation, verification and QA?
6	What is the expertise and depth of knowledge of the staff involved? What is the role of plant staff in the preparation and review of exercises?
7	Have automatic systems available been adapted for limitation of PTS? Are the procedures for the initiation of these systems available and adequate?
8	Have the time margins for the start-up of PTS/RPVI preserving equipment been properly determined?
3.1.2	**Status of symptom based emergency operating procedures (EOPs) related to PTS/RPVI, and of Severe Accident Management Guidelines (SAMGs) ….. treated under Section 6**
1	What is the current implementation status of those elements related to PTS/RPVI, which ones are to be used together with EOPs and SAMGs? Does the training currently provided treat this implementation as an issue?
3.1.3	**Administrative arrangements for personnel response**
1	Are there any PTS/RPVI challenges identified that can affect the CR personnel?
2	By what systematic means have limitations (of power supply, cooling media, etc.) associated with operating the equipment called for in the PTSA under accident conditions been identified and addressed?
3	How is it ensured that staff is sufficiently trained to fulfil all PTS/RPVI required functions?
3.2	**PROVISIONS FOR SAFE OPERATION**
3.2.1	**Overall concept of RPV integrity assurance and management provisions for RPVI/PTS**
1	Are there any organizational changes other than establishing the PTS Evaluation Group?
2	What is the staffing and qualifications of the PTS Evaluation Group within the TSC?
3	Have any administrative arrangements been made for the provision of required information to the PTS Evaluation Group during a severe accident?
3.2.2	**Operation provisions for sustaining RPVI**
1	What provisions and procedures have been implemented and/or changed in order to tackle RPVI related operation issues? When were those implemented? What is the current status?

No	VLI/VLI group description
3	**OPERATION & MAINTENANCE**
3.2.3	**Comparison with Western European state-of-the-art**
1	Are the provisions for RPVI sustaining purposes comparable with Western European state of the art?
2	Is all the relevant equipment subject to regular functional testing? What are the schedules for testing and/or inspections?
3.5	**MATERIAL EMBRITTLEMENT HISTORY VERIFICATION AND CONSEQUENCES**
3.5.3	**Internal and external reviews**
1	Have internal and external reviews been conducted of PTSA in order to establish comprehensiveness and suitability of mitigation precautions? Have the recommendations produced in the reviews been considered and implemented when considered an improvement?
2	Which periodical review process exists to check PTS/RPVI related measures for lessons learned elsewhere? Do the review mechanisms and the related administrative arrangements exist to identify potential changes in PTS mitigation and RPVI assurance?
3.5.4	**Organisation and conduct of PTSA validation tests**
1	What mechanisms and administrative arrangements have been in place to identify potential shortcomings of PTS mitigation and RPVI assurance, whenever identified?
2	What mechanisms and administrative arrangements have been in place to ensure effective and timely feedback from these reviews?
3.5.5	**Provisions for systematic revision of the PTSA**
1	What provisions are in place or are planned for updating and maintaining the PTSA as a “living document”? If a living document is not envisioned, what frequency of update to the PTSA is planned?
2	What systematic controls are in place or are planned to provide assurance that the PTSA is updated for maintaining it at the prevailing state-of-the-art as this changes over time?
3	Trough-out lifetime, are systematic PTSA revisions scheduled? At what intervals?
3.5.6	**Administrative arrangements for personnel response**
1	Have the training needs for different personnel involved in PTS management been systematically evaluated and documented? Which personnel will be trained this way?
2	Have the training programs and schedules for training, re-training, and testing of staff involved in PTS management been developed/documented?
3	Is there a deadline imposed by the regulatory authority for PTS management provisions implementation?
4	What personnel training management system is enacted, so that all persons involved in the decision making process have sufficient insights into PTS/RPVI, and in the potential consequences of their decisions? How is it ensured that staff is sufficiently trained to fulfil all PTS/RPVI required functions?
3.6	**EMBRITTLEMENT MANAGEMENT STRATEGIES**
3.6.7	**RPVI, PTS and qualifications of the staff**
1	Which qualification requirements have been defined for the staff involved into PTS/RPVI related operation?
2	Which qualification requirements have been defined for the staff involved into PTS/RPVI related technical support, ISI, NDT and implementation verification?
3	Which training program requirements have been defined for the staff involved into PTS/RPVI related technical support, ISI, NDT and implementation verification? What general timing rule applies to those training modules associated with PTS avoidance and mitigation as well as preservation of RPVI?

No	VLI/VLI group description
3	**OPERATION & MAINTENANCE**
3.6.8	**Training programme, training conduct, and training records**
1	What are the provisions for obtaining background, plant-specific information to support selection and implementation of PTS avoidance strategies? To what extent this information is provided by computerised information systems?
2	Are the criteria for checking avoidance by actions' success clearly defined?

Areas of Monitoring

No	VLI/VLI group description
4	**QA, FEED-BACK, CORRECTIVE ACTION**
4.1	**IMPLEMENTATION, MAINTENANCE AND THEIR VERIFICATION**
4.1.1	**ETE – PTSA implementation and verification**
1	What are requirements defined for the PTS/RPVI related implementation of provisions for recurrent maintenance, ISI, NDT and their verification?
4.1.2	**Surveillance program in ETE-1 and ETE-2**
1	Have there been maintenance or verification activities with respect to the surveillance programs in ETE-1 or ETE-2?
4.1.3	**Status of neutron irradiation minimisation in ETE-1 and ETE-2**
1	Which measures are envisaged to modify ETE-1 and ETE-2 core arrangements in order to minimise RPV wall irradiation?
2	Are measures are envisaged to change core neutron leakage? Which kinds of measures have been taken into consideration? What is their effect on the fuel design, enrichment, absorbing material, geometrical arrangement etc.? What are consequences for the power rating of the core? Which effect is expected for the safety margins for operation, e.g. DNBR, how have such effects been analyzed, and where are the related results documented?
4.2	**RISK INFORMED MODIFICATION, FEED-BACK ON OPERATION**
4.2.1	**ETE – PTSA implementation and verification**
1	What process has been undertaken to ensure that the technical basis for the Temelín PTSA is based on up-to-date insights into such accident sequences?
2	What are estimates about contribution to the frequency of severe accidents at Temelín from PTS sequences?
3	What are the overall results on accident progression calculations performed in support of the technical basis for the Temelín PTSA? Were the PTS evaluations performed with and without operator actions, and if so, what are the documented evaluations' results?
4	What concept has been employed in support of PTS assessment to identify plant vulnerabilities at Temelín? What were the vulnerabilities identified for Temelín?
5	With respect to PTS, what are the significant differences in accident progression timing between an intermediate PCL-leak and a double-ended guillotine rupture of an 850 mm diameter pipe?
6	With respect to PTS, in which of the sequences analyzed station blackout at Temelín was considered part of the events?
7	What are the criteria for applying Temelín EOPs/SAMGs specific to PTS avoidance?
8	What modelling concept has been applied to the determination of the mechanical integrity of reactor pressure vessel structures at Temelín for scenarios when re-pressurisation does occur?

No	VLI/VLI group description
4	**QA, FEED-BACK, CORRECTIVE ACTION**
9	How are the timing limits on reclosure/repressurization reflected in the Temelín PTSA?
4.2.2	**Surveillance program in ETE-1 and ETE-2**
1	Describe the Temelín surveillance program adjustment options for the timing of surveillance actions?
2	What results would induce modifications in the surveillance programs of ETE-1 and ETE-2?
4.2.3	**Status of neutron irradiation minimisation in ETE-1 and ETE-2**
1	How are uncertainties involved in irradiation minimization handled in analysis and in the surveillance program and PTSA? Which provisions are made for corrective action, if required?
4.3	**ISSUES RELATED TO QA**
4.3.1	**QA aspects for PTSA**
1	Is there a formal QA program applicable for PTSA?
2	What arrangements are in place to ensure that the PTSA are traceable and can be reviewed?
3	How were the organizations qualified for performing the PTSA? What were qualification requirements for the personnel involved in the PTSA?
4	Have any independent reviews of the PTSA been conducted?
4.3.2	**QA Program for the PTS design and manufacturing**
1	Were the QA systems, as used by equipment manufacturers, subject to verification?
2	Has the operational QA program at the plant been subject to independent peer review? Who performed the review? Have the recommendations been implemented?
3	Is there a deadline imposed by the regulatory authority for PTS management provisions implementation?
4	Have validation exercises been conducted? If not, what arrangements are planned?
5	Was the validation exercise properly designed in order to verify the completeness and adequacy of the PTS mitigation operation guidelines? Which accident scenarios were selected for the validation exercise(s)?
4.3.3	**QA Program for the PTS prevention and mitigation procedures**
1	Is there a QA program in place for PTS prevention and PTS mitigative procedures, including development, implementation, revision and experience feedback?
4.3.4	**QA Program for operation and operation feedback and operation feedback**
1	Has a QA program been developed and implemented for operational experience feedback analysis on PTS and RPVI?
2	Can all the lessons from PTS in operation and PTSA update exercises be properly analysed and used to propose improvements of the guidelines?

7.1 QA procedures (limited to considerations presented during the workshop)

7.1.1 Description of the issue – fundamentals

The purpose of quality assurance is to ensure in a verifiable way that the quality requirements with respect to the product forms, components and systems are established and, taking the respective load conditions into account that these requirements are met to the required extent during fabrication, assembly and during operation and maintenance until decommissioning. The quality requirements can only be planned, fulfilled and the fulfilment verified, if during any step the tasks are carried out with technical knowledge and under consideration of the Code requirements for design, construction, operation and maintenance. The goals of quality planning should therefore be to ensure that the protection goals stipulated in laws are reached.

The quality assurance as applied to the implementation process of procedures as well as to management procedures is of the same basic requirements, structure and result including feedback and corrective action control. In both cases, the key function is to verify and persist in sustaining the required safety level of the entire system, of its operation and of its supervision.

The individual quality assurance measures are part of a complete quality assurance program by which the fulfilment of all requirements can be verified and the gained experience fed back into the planning.

The safety standard establishes the basic requirements for quality assurance with the scope to specify requirements regarding planning, organization, technical and organizational procedures, documentation, test and inspections. The procedures should aim to prevent later occurrence of mistakes and failures.

These general requirements include any quality assurance tasks related to RPVI activities.

State-of-the-art requirements and regulations

The IAEA Guidelines [IAEA 1997] define the overall quality requirements for the demonstration of the RPVI using PTS analysis by the following principles:

(a) *“Selection of initial data and boundary conditions, computer codes and users, influence the quality of results, therefore all of them should be subject to quality assurance procedures.*

(b) *Any activity should be performed only by qualified personnel. A record documenting the qualification should be maintained.*

(c) *The origin and version of computer codes used should be clearly documented and must be referenced in order to allow a meaningful evaluation of a specific accident analysis. Computer codes should be verified and validated for the relevant area of their application; verification and validation should be documented.*

(d) *All sources of primary plant data should be clearly referenced. Derivation of input data for the analysis from primary information should be documented in such way, which permits adequate control, review, check and verification. A form should be used which is suitable for reproduction, filing and retrieval. Notes should be sufficiently detailed such that a person technically qualified in the subject can review, understand and verify the results.*

(e) *It is advantageous to have one "master" input deck. All calculations should be done introducing the necessary changes (e.g. initial conditions, functioning of safety systems) with respect to this "master". All such changes should be documented so that it can be traced to the date in which improvements/error corrections have been done. Inputs should be designated in a way that permits later checking. Data permitting reconstruction of calculated results must be archived (including relevant parametric studies).*

(f) *For each case analyzed a sufficient description of input data, basic assumptions and process and control system operational features should be provided giving a possibility of a unique interpretation and reproducibility of the results. It is recommended to follow the same format for all cases analyzed.*

(g) *"User effects" should be reduced to minimum. This implies that guidelines should be developed at the institution performing the analyses, permitting each member of the staff to benefit from the experience accumulated in applying a given computer code. For the same reason, data transfer between computer codes should either be automatic or it must be assured that they are defined in an unequivocal way.*

(h) *Results should be presented in such quality and detail to allow the reviewer to check the fulfilment of all acceptance criteria and to understand properly all elements and in particular the interdisciplinary aspects (interfaces) of the PTS analysis. The same format for presentation of results for all cases under consideration is recommended. Results of analysis should be archived for a sufficiently long period of time.*

(i) *All calculations and analyses should be checked by a competent individual other than the author. The following methods may be used for checking the adequacy and correctness of calculations:*

Independent review and checking of calculations,

Comparison of the results with results of other methods, such as:

- *Simplified calculations*
- *Alternate calculation methodology.*
- *Other appropriate methods may be also used.*

The review process and all comments as well as deficiencies found by the reviewer should be adequately documented. Specifically it must be documented which parts of calculations and results have been checked and which methods have been adopted.

In response, the author should properly address all comments and remove all deficiencies to the satisfaction of the reviewer.

(j) *All input data for structural analysis (like RPV geometry, material properties etc.) should be documented according to the QA manual prepared for the PTS analysis."*

7.1.2 Current plant status

Workshop presentation: J. Žďárek: QA programme for analysis, assessment and related activities

According to the presentation during the Workshop [ZDAREK 2004a] the quality assurance programme is directly linked to the quality assurance programme of NRI[64] Řež based on the principles, methodology and QA assurance requirements established by

- ISO 9001 standard
- IAEA Code and Safety Guides
- Regulation of the SÚJB No. 214/1997 Coll. on Quality Assurance in Activities Related to the Utilisation of Nuclear Energy and in Radiation Activities.

[64] Nuclear Research Institute Řež

With respect to the quality assurance procedures for computational analyses the following procedures were defined:

1. Design control of computational analyses
2. Contract review (determination of requirements related to the project, review of requirements related to the project)
3. Subcontracting conclusion
4. Procurement of computer tools (hardware, software)
5. Software validation and verification
6. Training of personnel performing computational analyses
7. Input deck compilation and preparation
8. Process control (documentation, record-keeping, output document filling, output traceability and retrievability)
9. Software management (ensuring unauthorised access is not allowed, ensuring unauthorised changes are not allowed, ensuring all version are properly stored)
10. Control of nonconformies (corrective action, preventive action)
11. Control of records (standard quality records, specific records)

7.1.3 Evaluation

The presentation covered only the very general description of quality assurance procedures for computational analyses. A thorough evaluation of the quality assurance programme for RPVI activities is not possible without information that is more detailed.

The PTS analyses are performed in NRI Řež – the NRI quality assurance activities were applied to the PTS analyses for the NPP Temelín.

It is not clear whether the operator has implemented a complete quality assurance programme that covers the RPVI activities globally.

The Austrian Experts were informed during the 2001 Workshop in Řež (26./27.02.2001) that the Regulation on quality assurance 214/1997 was supposed to be fully implemented after a transition period of 5 years.

According to Mr. Tendera, an IQA (individual qualification assurance) programme has to be performed for each NPP. The respective document covering the IQA for ETE was not available for the Austrian Expert team.

7.2 Conclusions about QA procedures

- Due to the unavailability of detailed information, it is not possible to judge the efficiency of quality assurance programmes related to RPV activities at NPP Temelín. However, the PN4 project has to be consulted with in order to appreciate QA accomplishments achieved in this very context.

With respect to the training program in the NPP Temelín and ETE RPVI/PTS management procedures:

- Verification and consolidation of a sound understanding of the actual RPV and plant systems situation requires procedures and management structures to be set up. This management should be set up for a process that is supposed to last for the entire plant life. The related prerequisites have been set-up in adequate proportions.

7.3 Issues of further interest, monitoring items for QA procedures

It is recommended to consider the questions concerning quality assurance (the "individual qualification assurance programme" for NPP Temelín) and training programmes for further monitoring.

8 POSITION OF THE SÚJB

Areas of Monitoring

No	VLI/VLI group description
4	**QA, FEED-BACK, CORRECTIVE ACTION**
4.4	**REGULATORY BODY: PTSA, RPVI, SURVEILLANCE, LICENSING AND EXTENSION**
4.4.1	**REGULATORY BODY'S POSITION AND LICENSING ISSUES concerning RPVI and PTSA National requirements**
1	Which national codes or regulations are the bases for the Regulatory Body's licensing procedures concerning PTSA?
2	Which national codes or regulations are the bases for the Regulatory Body's licensing procedures concerning the definition of operational pressure-temperature limits?
3	Which national codes or regulations are the bases for the Regulatory Body's licensing procedures concerning surveillance programs?
4	Which national codes, regulations and acceptance criteria are basis for the Regulatory Body's licensing of RPVI/NDT programs?
4.4.2	**Regulatory body's position on PTS and RPVI implementation**
1	Is there a deadline imposed by the regulatory authority for PTS management provisions implementation?
2	What are requirements by the regulatory body regarding the power rating of the core? What is the position on changes of the safety margins for operation, e.g. DNBR and which analyses requirements of such changes related to PTS avoidance and mitigation have to be satisfied by the operator and have the related results to be documented with the FSAR?
3	Did the Regulatory Body accept the Westinghouse concept before its implementation?
4.4.3	**Regulatory authority's requirements for surveillance programs**
1	Which requirements have been imposed to the ETE surveillance program by the Regulatory authority?
2	What are the SÚJB's regulatory requirements concerning the qualification, responsibilities, authority, and training of Technical Support Centre personnel applicable to the Temelín facility? What requirements exist for periodic re-qualification and refresher training?
3	Which set of safety factors was used/were determined to be applicable to the lower bound fracture toughness curve representative for the ETE RPV-steels? (I.e. factors n_K and na in the formula: $n_K.KJ(T, n_a .a) \le K_{IC}(T + \Delta T)$ …. (where a is the depth of the defect).

8.1 SÚJB position (limited to considerations presented during the workshop)

8.1.1 Description of the issue – fundamentals

The general fundamentals of nuclear safety regulation are summarized in the document of the NRA [NRA 2003]. They provide an adequate description of the issues addressed in this chapter and reflect the obligations and requirements to be fulfilled by a nuclear regulatory authority. These fundamentals are the evaluation basis for the Experts' Team for the information from and the experience with the Czech Nuclear Authority SÚJB including information received at the Specialists' Workshop in relation to its position.

Fundamentals of Nuclear Safety Regulation

The ultimate goal of nuclear safety is to protect the public and the environment from harmful effects of radiation. The international community agrees that the prime responsibility for the safety of a nuclear installation rests with the licensee, i.e. the organisation responsible for and in a day-to-day control of operations on site. The Nuclear Safety Regulator is responsible for oversight and the establishment of basic safety principles, the development of regulations as well as the enforcement of laws and regulations and for granting licenses. In addition, the Regulator has to actively communicate with the stakeholders on nuclear safety.

The regulator, therefore, needs:

- Effective political, legislative and financial independence
- Competent, motivated and sufficient number of staff
- Any external support to the nuclear regulator should be well qualified and
- Independent
- Sound regulatory processes supporting effective oversight and transparency as well as the safety responsibility of the licensees
- Support systems and tools to foster effectiveness and efficiency, such as quality and
- Competence management systems
- Active research programme to support regulation
- Powers to issue enforcement measures to effectively implement the mandate to protect public and environment
- Public communication strategy and tools
- Strategy for international co-operation including the exchange of experience and benchmarking.

As will be seen in the following chapter “Current Status” not all topics of the above summary of NRA duties were addressed during the workshop. Nevertheless, essential topics as an independent assessment, etc. were explicitly touched, some others implicitly.

A similar statement has to be made for the questions below, prepared by the Experts` Team on basis if the VLIs to be monitored at the Specialists’ Workshop.

8.1.2 Current status

To some extent, information about the position of the Czech Regulatory Authority SÚJB related to the RPV Integrity and PTS analyses for Temelín NPPs became available directly via two presentations of a SÚJB representative at the workshop in Prague. Additional information could be gained from the representations of the TSO UJV and the utility ETE. The discussions and answering of questions of the Experts’ Team after each Czech presentation provided for the Experts’ Team evaluation also valuable contributions and insight at the workshop how SÚJB appears to be able to manage its mandate as regulatory authority in the context of the RPVI and PTS issues.

To some extent, the Experts Team received documented evidence of the information provided.

Information gained from SÚJB Workshop presentation: M. Šváb: SÚJB Comment on Current Legislation Basis Aspects for RPV PTS:

The current licensing procedure related to the topic RPVI and PTS is based on the Atomic Act, 18/1997 Col. Articles 17 and 18, regulations, instructions and recommendations of the Czech State Office for Nuclear Safety, considered by SÚJB to be entirely based on IAEA recommendations and an acceptance document for the VERLIFE procedure by the SÚJB.

- Atomic Act, 18/1997 Col. Article 17 describes the "General Obligation of the Licensee" to fulfil the topic of RPVI and PTS assessment in a "systematic and comprehensive" manner according state of the art requirements, and to ensure that assessment results are put into practice.
- Atomic Act, 18/1997 Col. Article 18 describes the "Obligation from the Aspect of Nuclear Safety, Radiation Protection, Physical Protection and Emergency Preparedness", to "monitor, measure evaluate, verify, and record values parameters and facts related to the aspects addressed according procedural regulations.
- Regulation 214/1997 Col. comprises the QA in activities related to utilization of Nuclear Energy and provide criteria for categorization of classified equipment generally and provides in Article 22 specifications to the process for activities during operation of the NPP in relation to "operation and shut down control", the incidents and accidents "notification and evaluation" and the control of experiments and tests.
- The instructions and recommendations for lifetime assessment of VVER RPV and reactor internals during NPP operation [SÚJB 1998].
- Section IV if the Standard Technical Documentation of Association of Mechanical Engineers ASI of the Czech Republic. Residual lifetime assessment of VVER nuclear power plants components and piping, edition dated 1998).
- IAEA Guidelines on PTSA for WWER nuclear power plants [IAEA 1997].
- VERLIFE acceptance document issued by SÚJB.

During the discussion phase of this topic it was stated by the SÚJB representative that the guidelines referred as basis for the licensing procedure are not mandatory for the licensee but following these guidelines by the licensee "would be fine".

SÚJB identified no specific plans to have an independent review of the PTSA related RPVI work accomplished. It rather mentioned in the Czech presentation related to SÚJB subcontractors' involvement in monitoring and reviewing the ETE PTS activities and results, that the Regulatory Authority did obviously not involve any specific independent subcontractor e.g. GRS or any other TSO for assessment of the PTSA material produced by the utility ETE. However, SÚJB mentioned in this context that the developing entity of the VERLIFE (M. Brumovský, TSO UJV, et al.) had involved several institutions of this kind for independent assessment and the Regulatory Authority refers to national institutes, universities and other organisations and to the State Energy Inspectorate (SEI) as well as eventually the Trenčianskej Univerzity Alexandra Dubčeka in Trenčín, Slovakia and to the PHARE projects which touch somehow the related issues *"like an independent auditory or tutor in this field"*. Anyhow, SÚJB's involvement would be limited already by the workforce of three persons that are in charge of reviewing and assessing the mechanical part of the PTSA.

Obviously, no specific deadline for PTS management provisions' implementation was fixed by SÚJB. The Czech side referred in this context to a comment by the IAEA that there is no need of implementation of PTS measures within the first 5 years of operation.

In the Specialists' Workshop's final SÚJB presentation "Requirements on PTS" several topics of the above questions were addressed generally. OKB Gidropress for initial assessment had already made generic calculations of the PTS regimes incurred during RPV lifetime. According to SÚJB`s viewpoint, ETE has performed additional assessments and calculated *"the most conservative PTS regimes"*.

Reported in a general manner was about:

- Existing R&D projects related to an experimental program on "Disbanding" Defects Tape Influence Assessment of RPV Integrity (Cladding manufacturing deficiencies related);
- Sampling of RPV Inner Wall Cladding for Neutron Fluence Assessment and
- Probabilistic Assessment of RPV Fast Failure.

The following planned R&D projects were mentioned in the presentation:

- Determination of Real Progress of Radiation Damage on the RPV Wall;
- Influence on Warm Prestressing to Integrity of RPV during PTS Events.

Reference was made in the SÚJB presentation to three PTS activities in the frame of PHARE Projects, which are followed by the SÚJB. They comprise:

- Re-assessment of the RPV Internal Stress State Based on Real Service Irradiated and Derived Mechanical Properties;
- WWER cladded RPV Integrity Evaluation (with respect to PTS events).

The Regulatory Authority provided some general information on the Specialists' Workshop about inspections performed by SÚJB. It related to the:

- Surveillance program;
- ISI during outages;
- Aging management;
- Diagnostics.

The SÚJB conclusion was reported as follows: "*Based on combination of all the mentioned tools and results we can state that the PTS issues are properly treated at the Temelín NPP*".

The "low leakage core arrangement" of Temelín NPPs was licensed by SÚJB. SÚJB participation in the surveillance program will enable them to see, what is going on at the site. SÚJB representatives are not participating in making real technical expert assessment, but participate in ND inspections and are following all projects that are in progress.

From SÚJB it became clear that the process of standardisation of codes is followed as described by SÚJB at the PN7 Specialists' Workshop on Severe Accidents in Prague (see presentation "Regulatory Approach to Accident Management").

The SÚJB position and obligations in the frame of the PTS issue became evident not only through the SÚJB presentations but also through the presentations of representatives of ETE and UJV. In the following reference is made to these presentations and the related information concerning SÚJB.

All ISI qualification activities are performed under the supervision of the Regulatory Body. The manufacturing process (of the RPV) complied with the Individual Quality Assurance Programmes approved by the Regulatory Authority.

Related to the RPV NDE Programme it is stated: "*Separate individual Quality Assurance Programmes were prepared and subsequently approved by the regulatory authorities for all types of RPV inspections.*"

The confirmed status of the VERLIFE Project was said to be, that the "*Procedures* (of VERLIFE) *have been accepted for lifetime and integrity evaluation of components and piping in WWER type NPPs by State Regulatory bodies in Czech Republic.*"

Little information was made available on the status of a consolidated QA process adopted ensure quality of the RPVI/PTSA applications, for the integrity models for example. Only some indications were given, e.g. that the SYSTUS code used for the calculations is periodically judged by the SÚJB authorizing sub-committee Nr. 5 (last time under the code number 517, on 2. July 2003). The surveillance QA of codes requires the approval of the computer codes, it is therefore mandatory to provide also validation information about them and updates as well. This is a standard procedures application in place for all computer codes related to calculations relevant to safety.

8.1.3 Evaluation

The SÚJB position on the "PTS requirements" implementation versus the licensee appears to indicate a rather vague and not too strong position in assuring the RPVI and PTS precautions fulfilment.

No clue was found about a deadline set to the licensee by the SÚJB for PTS requirements implementation.

Concerning the R&D Projects mentioned the Experts' Team concluded from the information provided that the existing and planned R&D projects are national ones.

The involvement of SÚJB in the existing R&D projects was not explicitly addressed but from the context it might be drawn, that SÚJB is monitoring all these activities. Evidence of results of all these projects and their influence on existing procedures for the Temelín NPPs and the resulting SÚJB requirements are of specific interest to follow-up activities.

SÚJB is evidently part of important international projects. The Experts' Team did not assure itself about SÚJBs engagement or involvement in these international projects. Nevertheless, the information given indicates that the State Regulatory Authority SÚJB has good knowledge about these projects and would have therefore good reason for its results to be applied in its licensing activities related to Temelín RPVI and PTS issues.

Based on the very general information from SÚJB in relation to an "independent assessment" and the limited own engagement in assessing the PTSA material produced by ETE the Experts' Team is not quite sure, that the Regulatory Authority is enabled by all means to perform an independent assessment of the ETE documents delivered.

SÚJB's way of dealing with "Inspections" gave an impression about the specific areas of SÚJB involvement in only a very general manner. SÚJB`s personnel capacity and support in this context should be specified in more detail.

The Experts' Team would welcome the substance of SÚJB's conclusion about the proper treatment of the PTS issues at the Temelín NPP be supported by more substantial information and some more evidence about the SÚJB capacity and capability in this area. In this context the procedure is of interest, how the SÚJB has accepted and independently approved the VERLIFE concept for the Temelín WWER 1000 reactors. SÚJB currently advocated position about independent assessment of the Temelín PTSA was interesting to the Experts' Team, which referred to the international participation in setting up the VERLIFE concept but did not explicitly refer to SÚJB`s specific activities in this field.

The Regulatory Body enjoys the support of a number of TSO organizations to accomplish its part in the evaluation of matters relating to the RPVI and PTS related activities, however the extensive work would require evidently substantial additional effort, since in more then one instance the answer to questions regarding the time schedule and finalization date were not named.

It was not reported in the Czech presentations to what extent SÚJB was involved in assessing and approving EOPS related to PTS prevention for Temelín NPPS and thus this is an open question.

The Experts' Team received good insight into the essential topic of external support and independence, which is addressed in the NRA fundamentals.

8.2 Conclusions regarding the Position of the SÚJB

- The SÚJB position on the “PTS requirements” implementation versus the licensee is an indication of their observing position in assuring the RPVI and PTS precautions fulfilment.
- From the information gained about SÚJB`s position the Experts’ Team states – in following the IAEA suggestion – that it is a valid aim to enhance SÚJB´s “strength”. Its personnel capacity and possibilities ought to be increased by all means appropriate and necessary to follow its obligations generally and specifically with respect to the RPVI and PTS issues at the Temelín NPP in a sufficient and effective manner.

8.3 Issues of further interest, monitoring items about the Position of the SÚJB

- It is recommended to monitor the status and results of the existing Czech R&D projects on PTS. Also some more details about scope and time schedule of the planned R&D projects are of specific interest. Evidence of results of all these projects and their influence on existing procedures for the Temelín NPPs and the resulting SÚJB requirements are of specific interest.
- It is also recommended to monitor and review the results of the international PHARE projects on PTS.
- It is recommended to ask SÚJB for information in detail about pending (remark: at the time of the workshop) PTS scenarios and results.

9 CONCLUSIONS

The conclusions compiled for the various aspects of RPVI and PTS are collected here for the sake of better appreciation and in order to enable the reader to see them in the broader context of monitoring results in a condensed manner.

9.1 Conclusions Summary

The demonstration of RPVI (reactor pressure vessel integrity) throughout service life is performed by the Czech Experts, for Temelín NPP, using the VERLIFE methodology. From a comparison to the Russian Code and the IAEA Guidelines, the Austrian Experts' Team has found that the VERLIFE methodology, as applied to the Temelín RPVs, makes use of reduced safety margins (i.e. reduction of the postulated crack size, elimination/reduction of safety factors, non-conservative assumptions for the fracture mechanics analyses). In combination with other uncertainties concerning material/embrittlement properties and apparent reductions of conservatisms in several respects, the Austrian Experts' Team considers the resulting global safety margin for the Temelín RPVs as not being sufficient.

The complete VERLIFE methodology requirements and their application to the Temelín NPP have not been available to the Austrian Expert's Team. For the applied VERLIFE methodology the Austrian Experts' Team had to rely essentially on the information provided during the Specialists´ Workshops.

The Austrian Experts' Team also found that the Czech approach – as presented – for PTS analyses is in accordance with the state of science and technology, with respect to the concept, the methodology and the applied computer codes. The most severe transients analysed are well comparable to those regarded as representative for WWER-1000 installations according to current knowledge. All accident groups important in a PTS analysis were considered.

However, a number of issues remain to be clarified:

- The basis for the analyses appears to be insufficient: Although all accident groups important in a PTSA were analysed, in some cases the time frame of the simulation might not have caught critical loads to the reactor pressure vessel, since simulation results were available only for the phase ending just before repressurization would take place. Within a number of accident groups, the transients analysed in some cases cannot be considered as the most critical ones. For some transients it is necessary that emergency operating procedures be performed within a narrow time window to avoid brittle failure of the RPV.
- There are apparent reductions of conservatisms. Some VERLIFE criteria are weaker than those required by the IAEA Guidelines. Applying the values concerning postulated crack sizes, safety factors, WPS (warm prestressing effect criterion) as required by the IAEA Guidelines would not result in the demonstration of RPVI requirements' fulfilment throughout lifetime.
- Uncertainties – procedural as well as intrinsic – identified regarding the PTS assessment for Temelín NPP concern, for example: TH transient models, mixing behaviour models, embrittlement behaviour of the RPV materials as well as initial materials' brittleness properties, fluence determination and the introduction of measures for fluence minimization, and areas of in-service-inspection (ISI), where qualification has not yet been achieved. These are further critical points remaining for clarification.

- Conservatism is further reduced by including the intact cladding zone as structural reinforcement into the Finite Element model, including non-conservative assumptions for fracture mechanics analyses at the cladding/ferritic steel interface (as confirmed by a pilot study of the Austrian Experts' Team). Accordingly, not all types of underclad cracks have been evaluated.

Regarding the surveillance program, which is monitoring embrittlement progress, in particular the location of the samples, it has to be pointed out, however, that it represents a considerable improvement compared to other WWER-1000s of the same vintage.

Consequently, the future exchange of information on RPVI and PTS should above all cover the following issues:

- Regarding PTS analyses, the consequences of additional critical conditions, and of an extended time frame for some of the sequences calculated, are of interest, as well as the consideration of all crack sizes and crack positions of relevance in fracture mechanics (including stability considerations).
- The progression of embrittlement and the remedies taken should be further observed. This includes surveillance results for both units of the Temelín NPP, in particular the results of samples with higher initial critical temperatures brittleness, irradiated in unit 2.
- The comparison of materials' characteristics determined within the qualification tests, the extended acceptance tests and the lifetime evaluation programme with the surveillance programme data is of interest, in order to evaluate the scatter of materials' properties.
- Embrittlement mitigations measures, in particular core configuration, refuelling pattern and enrichment changes, are of interest.

In the course of further information exchange, the issues listed here could be combined with the issues remaining for information exchange under Item 4 (Non Destructive Testing) of ANNEX I of the "Brussels Agreement", regarding the reliable detection of all PTS relevant defects.

Detailed conclusions

Benchmarking the presentations at the Specialists' Workshop against internationally accepted guidance, recommendations and ruling has led the Austrian Experts' Team to the following observations: Many of these observations are also based on generic calculations and investigations that were conducted while preparing the workshop.

- The Austrian Experts appreciate that the Czech side is no more considering the operational pressure-temperature limit curves as appropriate demonstration of avoidance of unacceptable PTS sequences.
- The RPVI concept, as it pertains to the PTS analysis approach, appears to follow the state-of-the-art practice and the IAEA Guidelines with respect to analytical methodology. The IAEA Guidelines safety precautions were significantly reduced the way they are interpreted in the new VERLIFE methodology.
- The presented Czech approach for PTS analyses (part of the VERLIFE) with respect to the concept, the methodology and the applied computer codes are considered to be in accordance with state-of-the-art procedures.
- Evidently all thermal hydraulic calculations work has been performed with state-of-technology computer codes, which were validated for WWER-1000 use. Once completed the RPVI/PTS related TH-analyses can be considered comprehensive. TH-analyses should provide a sound basis for the selection of candidate transients for the mixing and heat transfer calculations to be conducted subsequently. The use of assumptions, which are not conservative for the specific scope and represent therefore an impact on safety, should be reconsidered.

- The most severe transients are by all means comparable to those considered representative for WWER-1000 installations according to current knowledge. In some instances the time frame observed in the simulation might not have caught the essence of the loading to the RPV, since re-pressurization during the up-following accident-sequence might just not have taken place that early (i.e. before ceasing the simulation).

With regard to "Mixing Calculations and Heat Transfer" issues:

- The mixing calculations for the accident transients within the PTS analyses performed appear to be in accordance with the state-of-the-art in international practice and comparable to calculations for other WWER-1000 reactors.

With regard to FEM calculations and Fracture Mechanics evaluation:

- The applied computer codes for the FEM simulation and the consideration of elastic-plastic material behaviour is considered to be in accordance with the actual state-of-the-art. The PTS assessment can be considered a consolidated approach, up to now unprecedented for WWER-1000 reactors.
- The IAEA Guidelines allow the use of postulated crack depths shallower than the normally required ¼ of wall thickness (which is for the WWER-1000 about 50 [mm]) for the case of the NDT-Program enabling the safe detection of the respective small defect sizes. For this case the IAEA Guidelines require the mandatory use of safety factors: Safety factor 2 for the crack depth or safety factor $\sqrt{2}$ for the stress and ΔT = 10 [K] for the embrittlement induced shift of the critical brittle fracture temperature. In accordance with VERLIFE [PISTORA 2004a] the Czech Experts postulate a crack depth of 20 [mm] only ($^{1}/_{10}$ wall thickness, which is significantly smaller than ¼ wall thickness) but do not apply any of the safety factors. (e.g. as required according to the IAEA Guidelines).
- The Czech approach is also deviating from the IAEA Guidelines [IAEA 1997] with respect to the missing variations of the crack size and crack geometry. The following investigations have not been presented:
  - The analyses for very shallow cracks (a < 6 [mm]) and
  - Large cracks (a = 20 [mm] up to ¼ of the wall thickness) and
  - The variation of the aspect ratio to a:c = 1:10.
- The approach taken for integrating the cladding zone into the FE modelling introduces furthermore a reduction of conservatism, not only when excluding elliptical under-clad cracks, but also because assuming a Stress Intensity Factor (SIF) levelling out to SIF=0 exactly at the cladding/base-material interface does not correspond to reality. This has been reconfirmed by pilot case simulations conducted during the monitoring process.
- The FEM model represents one half of the reactor pressure vessel. This procedure does not include the stresses from the superposition of the cold plumes, the strain induced distortion of the cylinder and the interaction with the RPV bottom and the RPV head (deformation hindering). It should be noted, that this approach is in accordance with the international practice. The simulation using a mesh covering the complete RPV would represent an outstanding effort.

With respect to the PTSA:

- All accident groups important to be treated in a PTSA were analysed. For WWER-1000 reactors this is the first PTSA with a completeness not achieved up to now.
- The PTS loads for WWER-1000 are extremely high. For a postulated crack depth of only 20 [mm] the resulting T_k^a values are below 70 [°C] in four cases and 3 accident groups, no comparable behaviour is found with any other reactor types, e.g. WWER-440. This is a consequence of the very effective emergency coolant injection systems that are able to compensate large breaks up to ND 850 but induce at the same time a severe thermal shock load at the RPV wall.

- The lowest T_k^a values are found for small to intermediate break sizes, where in addition to the thermal shock load a full or partial re-pressurisation of the primary coolant circuit might occur .
- The operator must perform the appropriate emergency operation procedures (EOPs) at the correct moment in order to cope with several accident transients (PSV41) and at the same time avoid brittle failure of the RPV. However, it is not international practice to require "guaranteed" operational procedures of the personnel; therefore this must be considered a considerable reduction of conservatism in the handling of emergencies.
- Some accidents (PSV43) have not been calculated until to the point of applicability of the 90% WPS-criterion.
- In some cases the definition of the accident transients cannot be considered the most critical one: In the accident group PRZ SV the total loss of off-site power has not been included, although this is required by the IAEA Guidelines. Including total loss of off-site power would induce a re-pressurisation in the primary circuit following the re-closure of the pressuriser safety valve.

With respect to the safety factors required by the IAEA Guidelines it has to be stated:

- The VERLIFE methodology as applied by the Czech Experts for the Temelín NPP uses only postulated crack depths of 20 [mm] ($^1/_{10}$ wall thickness, which is significantly smaller than ¼ wall thickness) and no safety factors, which is not in line with the pertinent IAEA Guidelines.
- The VERLIFE methodology as applied by the Czech Experts for NPP Temelín is applying the 90% WPS criterion although the IAEA Guidelines recommend the 80% level, if applied at all. This modification significantly reduces further conservatism, which violates the need to compensate for uncertainties in embrittlement prediction for radiation-sensitive RPV steels.
- Even though applicability of the WPS effect is still judged controversial in the international community due to theoretical and experimental uncertainties[65] it is applied for the Temelín RPV integrity.
- The consequent application of the IAEA guidelines would lead to a different assessment result than advocated by the operator of Temelín, e.g. in cases where the 80% WPS-criterion, together with the safety factor $\sqrt{2}$ and the required safety factor $\Delta T = 10$ [K] should be applied.

With respect to the surveillance program in the NPP Temelín and ETE RPV material embrittlement:

- The use of optimised steels – not radiation embrittlement susceptible/sensitive -, one basic element of state-of-the-art RPV integrity, is not met for the barrier – the Temelín RPVs.
- The modified surveillance program in the NPP Temelín allows the determination of reliable embrittlement data with respect to irradiation temperature and neutron flux/fluence at the samples irradiation location.
- The modified surveillance program causes inaccessibility of RPV wall in the container area and therefore for NDT in regions close to weldment 4, the active core and core zone.
- The evaluation of published surveillance results from WWER-1000 materials taking into account the estimated irradiation temperatures does result in considerable uncertainties about the neutron embrittlement of WWER-1000 steel. It is therefore obvious that the specification in the Russian Code (A_F = 20 for welds, 23 for base material) cannot be considered conservative.

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[65] This is the case especially with respect to the real situation in the component and the temperature/pressure history during a realistic PTS event.

- Although the first reliable results ever regarding a WWER-1000 will be available from the Temelín surveillance program, uncertainties about the WWER-1000 RPV steel embrittlement persist: the RPV specific surveillance program cannot provide a reliable statistics background for the prediction of the material degradation, since every set of samples withdrawn and evaluated provides for only one single data point to be added to the irradiation embrittlement versus time correlation.
- The embrittlement coefficients determined so far for Temelín specific materials are based on irradiation in test reactors with high lead factors. The existing dose rate effect might have adversely affected the embrittlement and the coefficients determined, i.e. the embrittlement might in reality be higher than measured.
- The material properties data in the passports indicate that the initial critical temperature of brittleness T_{k0} can vary by tens of degrees from one weld metal charge to another. It has not been possible to check whether the temperature margin δT_M (10 [K] for the base material and 16 [K] for the weld metals) as defined within the VERLIFE methodology, in order to cover the scatter of the mechanical property values, have been taken into account for T_{k0} assessment.
- This fact and the uncertainties of the specified embrittlement coefficients need to be taken into consideration by using the safety factor ΔT as required by the IAEA Guidelines [IAEA 1997].
- Weld no. 4 in ETE-1 was welded with two different electrode heat charges (Sv12Ch2N2MAA, heat number 17084 and 170007) for both heat numbers surveillance samples were fabricated; the surveillance program of ETE-1 is performed using the samples welded with the same electrode heat than weld no.3 (T_{k0} = -50 [°C]). The other weld metal with T_{k0} = –30 [°C] will be irradiated within the surveillance program of ETE-2. In view of the Austrian specialists this is a shortcoming because the results on irradiation embrittlement for the weld material with the highest T_{k0} of ETE-1 will not be available without significant delay.
- The fracture toughness curve formula used in the VERLIFE methodology can be considered conservative as compared with fracture toughness curves of other National Codes.

With respect to the NDT/ISI program performed in NPP Temelín:

- The ISI using ultrasonic NDT methods for the RPV cylindrical wall has successfully been qualified. The methods can as such be regarded to basically enable detection of all kinds of crack-like defects, which are of special concern for the PTS events and their analyses, e.g. cracks close to the claddings' interface to the base material layer with an $^a/_c$ aspect-ratio of e.g. 0,3 and with different depth, depending on the PTSA postulated defects. A semi-elliptical crack seems to be the most critical case for NDT, which starts at the cladding interface and extends 8 [mm] deep into the ferritic wall. Although qualification using the RPV wall test block demonstrated the basic potential of the applied UT methods to allow detection of those defects, some problems are not yet finally solved.
- The test block does not contain the cladding condition at the welds and on its vicinity, where one has to take into account a considerably higher noise level and therefore a higher false call rate, as mentioned in the qualification report. This requires special countermeasures, e.g. additional Eddy Current Testing (ECT) in areas with an elevated number of UT indications. This is particularly needed, because the VERLIFE concept requires an intact cladding, especially at locations of near cladding cracks in the ferritic wall. The remaining ligament between the crack tip and the wet inner surface can only be proven with appropriately qualified ECT methods. The safety evaluation regarding the absence of flaws important to PTS has not been finalized up to now, since neither the qualification of, nor the inspection with the ECT method as required has been carried out yet.

- Two more ISI areas bearing specific PTS concerns are the inner corner of the inlet nozzles and the welds connecting the primary loop to the RPV. For both areas qualification exercises have been announced, but have not been finished yet and presented. Of special interest are the PTS relevant crack sizes within the nozzle corner and the connecting weld, in order to judge the difficulties the NDT techniques will have to guarantee sufficient detectability and a reasonable false call rate.
- In view of the not yet finished remaining NDT activities, but needed to prove the absence of all kind of PTS relevant cracks, one must conclude, that the NDT inspections carried out until today cover only in part all the ISIs required. According to the information given at the PN9 Workshop, completion of the ISI concerning the PTS analysis is in preparation, with several qualification activities ongoing, but will certainly not be reached before the foreseeable next RPV ISI.

With regard to Core Design and Radiation Embrittlement Mitigation:

- The OUT-IN strategy is a well-known early means of embrittlement mitigation; the ETE specific information contained in the presentation did not give a clue to the question, whether introduction is made for irradiation embrittlement mitigation, or just as a side effect of power output optimization. The PTS relevant effects of the RPV fluence reduction management can be derived from the fluence distribution only. Nevertheless, information presented was limited to power distribution sketches.
- The statement during the Specialists' Workshop, that operation will take place well below fluence calculation input, does not per se endorse that embrittlement is managed properly. The RPV fluence reduction management policy is one element to be enacted along with plant operation.
- The Westinghouse core design used in a WWER-1000 reactor is the first of its kind to be validated. Apparently, the core design has not yet been modified aiming to a fluence minimization at the reactor pressure vessel wall in order to reduce the neutron embrittlement of the steel. This improvement is envisaged be implemented at one of the upcoming refuelling outages of the core. Up to now the intended changes have not been presented.

With regard to EOPs and SAMGs transition:

- Extensive feedback from plant analyses was used to more appropriately adapt the EOPs outline and elements to an up-to-date emergency management tool. It can be understood from the overview presentation, that the concept is suitable for proper adaptation. This work is evidently a successfully ongoing process.
- The EOPs as well as the SAMGs and associated measures are well in line with the state of science and technology requirements, given the equipment to be used be qualified or been qualified for the intended use in the respective operational regime.

Conclusions concerning the issue of quality assurance and training:

- Due to the unavailability of detailed information it is not possible to judge the efficiency of quality assurance programmes related to RPVI activities at NPP Temelín. In any case, together with the evaluation of quality assurance the improvements achieved for QA are appreciated.
- Verification and consolidation of a sound understanding of the actual RPV and plant systems situation requires procedures and management structures to be set up. This management should be set up for a process that is supposed to last for the entire plant life. The related prerequisites have been set-up in adequate proportions.
- The training and implementation activities are comprehensive and compare well with activities in other NPPs in Europe. In some instances thoroughness was most probably given precedence before timeliness when implementing EOPs training opportunities.

Conclusions concerning the SÚJB position:

- The SÚJB position on the “PTS requirements” implementation versus the licensee is an indication of their observing position in assuring the RPVI and PTS precautions fulfilment.
- In line with the IAEA IRRT Mission recommendation the Experts’ Team considers that it is a valid aim to enhance SÚJB´s “strength”. Its personnel capacity and possibilities ought to be increased by all means appropriate and necessary also in the RPVI and PTS context.

9.2 Detailed issues of further interest and Monitoring Items

The team of Experts recommends pursuing topics of high priority in the framework of the pertinent Bilateral Agreement between the Federal Republic of Austria and the Czech Republic. This concerns the implementation and results from the RPVI Program, VERLIFE and the related PTSA. In addition, since the ongoing RPVI/PTS information exchange process is supposed to be continued for the entire plant life, it is recommended to follow plant operation by continuous exchange of information.

Since the present RPVI work did not explicitly take into consideration cold over-pressurisation and outage-issues, no comments will be found here on these topics.

These items recommended are as follows:

- The consideration of additional critical conditions, such as total loss of off-site power,
- The time frame of sequences calculated – some transient have not been performed for a time sufficiently long, so that an over-pressurisation during the subsequent accident transient could not be considered -, and
- The consideration of fracture mechanics regarding all the crack sizes and crack positions of relevance, and stability considerations (smaller cracks might grow and become instable during the up following transient sequences).
- The embrittlement progression as well as the remedies taken and the actual RPVI verification and consequences.
- The comparison of the materials characteristics determined within the qualification tests, the extended acceptance tests and the lifetime evaluation programme cited during the Workshop [BRUMOVSKY 2004a] with the surveillance programme data in order to evaluate the scatter of materials characteristics.
- The information on the results of the surveillance programme for both units. Special emphasis should be dedicated to the surveillance results of the weld no.4 samples (including the heat affected zone). The first results of the surveillance capsule removed in May 2004, will be available in 2005.
- The information on the results of the surveillance samples irradiated in unit 2 (esp. specimens of weld no.4/unit-1 and weld no.4/unit2, including HAZ) should be included in the future information exchange with special emphasis during the next years. At the same time it would be desirable to obtain information whether specimen of weld number 2 are included in the PTS considerations.
- Continuous information on the experimental assessment evaluation of the neutron embrittlement of ETE materials using surveillance specimens in order to confirm the application of temperature margins as defined in the VERLIFE methodology (upper boundary of the radiation induced T_k shifts to be used in the RPV lifetime evaluation).
- The Temelín RPV embrittlement mitigation is of utmost importance for RPVI; therefore fuel-reload as well as reload-pattern changes are envisaged after one of the next campaigns. The information provided up to now is coarse; it stipulates further interest.

Future information exchange should also include:

- Main coolant recirculation line penetrations,
- Vessel head control rod penetrations,
- Core instrumentation and other service penetrations,
- Main flanges' tightness, and
- Major environmental and other damage mechanisms contributing to the loss of integrity, like main coolant chemistry, hydrogen diffusion, corrosion, load cycling, severe accident behaviour, as well as integrity preservation and surveillance measures ascertaining LBB applicability and leakage detection instrumentation,
- The damage progression as well as the remedies taken and the actual RPVI verification and consequences,

since the Workshop did not cover those RPVI relevant issues.

Concluding statement:

The Czech Experts make use of the VERLIFE methodology for demonstrating RPVI (reactor pressure vessel integrity) throughout service life of the Temelín RPVs. Compared to the Russian Code and the IAEA Guidelines the VERLIFE methodology has reduced the safety margins, adopted via inherent methodologies like the reduction of the postulated crack size, reduction of safety factors, the non-conservative fracture mechanics assumptions etc.

In combination with other uncertainties, such as modelling of TH transients, mixing behaviour modelling assumptions, material and embrittlement properties, fluence determination, NDE reliability, etc., the resulting global safety margin cannot be considered sufficient. Therefore the Austrian Experts' Team recommends continuing to follow up on those items, relevant for the completion of the VERLIFE methodology:

- Additional PTS analyses and their upgrading
- Surveillance specimen evaluation (of both units)
- Integrity verification dedicated NDE program
- Progress in embrittlement mitigation

The update of the Temelín RPVI demonstration specification based on the VERLIFE methodology would also be of high priority.

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ABBREVIATIONS

15Ch2MFA 15Kh2NMFA-A Sv12Ch2N2MAA West type 10MnMoNi5-517 MoV8-4	Heat resistant low-alloy steels with primarily tempered bainitic microstructures quenched and tempered. (Typical compositions are C(0,05÷0,2%), Mn(0,7÷1,6%), Mo(0,4÷0,6%), Ni(0,2÷1,4%), Si(0,2÷0,6%), and Cr(0,05÷0,5%)).
3D	Three-Dimensional
3SGT	3 Steam Generator Tube rupture
[Item No.1]	Identification chosen by the AQG/WPNS for the RPVI/PTSA Roadmap
AASSL	Actuation of Automatic Systematic Load
AC	Alternating Current
AFWS	Auxiliary feed water system
AQG/WPNS	Atomic Questions Group/Working Party on Nuclear Safety of the EU
ARCS	ARCS Seibersdorf Research GmbH
ASI	Czech Association of Mechanical Engineers
ASME	American Society of Mechanical Engineers
ATHLET	TH code (GRS maintained)
BMU	Bundesministerium für Umweltschutz und Reaktorsicherheit – German Federal Ministry for the Environment, Nature Conservation, and Nuclear Safety
BNFL	British Nuclear Fuels Limited – parent company of Westinghouse
BOL	Begin of life
BRU-A s	Steam generator safety valve, atmospheric relief
BRU-K	Steam generator safety valve, turbine bypass
ČEZ, a.s.	Check utility – joint stock company ČEZ, a.s.
ČEZ-ETE	ČEZ-Elektrarne Temelín – the portion of ČEZ, a.s. Operator of ETE
CFD	Computational Fluid Dynamics
CL	Cold leg
CL1	Cold Leg 1
CRP	Copper rich precipitate
CRP	(Co-ordinated Research Program of IAEA)
CSF	Critical Safety Function
DBA	Design Basis Accident
DBTT	Ductile-brittle transition temperature
DC	Downcomer
DEGB	Double-ended guillotine break
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
ECS	Emergency Cooling System

EOL	End of Life
EOP	Emergency Operating Procedure
ETE	NPP Temelín
EU	European Union
EURATOM	European Atomic Energy Community
FE	Finite element
FEM	Finite Element Method
FRAME	Fracture Mechanics Based Embrittlement
Gostekhnadzor	Russian TSO
H_2	Hydrogen
H850max	Large break LOCA in hot leg (diameter 850)
HA	Hydro accumulators
HAZ	Heat Affected Zone
HHSI	High Head Safety Injection
HHPIS	High Head Pressure Injection System
HL	Hot leg
HL1	Hot Leg 1
HPI	High Pressure Injection
HPIS	High Pressure Injection System
HTC	Heat Transfer Coefficient
HZP	Hot Zero Power
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IAEA NUSS	See NUSS
IAEA SG	IAEA Safety Guide
IC	Inside containment
IE	Initiating Event
INSAG	International Nuclear Safety Advisory Group (IAEA)
IRF	Institut für Risikoforschung
IRR	Institute of Risk Research, University of Vienna
ISI	In-Service Inspection
KB 190	Test sample for NDT qualification
KTA	Kerntechnischer Ausschuss – German Nuclear Standards Commission
LBB	Leak-before-break
LEFM	Linear Elastic Fracture Mechanics
ligament	Net Non-Cracked part of cross-section
LOCA	Loss of Coolant Accident
LOFA	Loss of Flow Accident
LOOP	Loss Of Offsite Powers

LPIS	Low Pressure Injection System
MC Techniques	Monte-Carlo Techniques
MCP	Main Circulation Pump
MIV	Main Isolation Valves (Turbine Stop Valves)
MNP	Manganese-Nickel Rich Precipitate
MSH	Main Steam Header
MSHR	Main Steam Header Rupture
MSIV	Main Steam Line Isolation Valve
MSLB	Main Steam Line Break
ND	Nominal diameter
NDE	Non-Destructive Evaluation
NDT	Non-Destructive Testing
NEA	Nuclear Energy Agency (OECD)
Ni	Nickel
N_{nom}	Nominal Reactor Power
NPP	Nuclear Power Plant
NRA	Nuclear regulatory Authority
NRC	Nuclear Regulatory Commission
NRI Řež	Nuclear Research Institute Řež
NUREG	Nuclear Regulatory Series of NRC
NUSS	IAEA Nuclear Safety Standards
OC	Outside containment
OCA-P	Russian Code
OECD	Organisation for Economic Co-operation and Development
OPB	Russian safety standards
PCS	Primary Coolant System
PIA	Post-irradiation annealing
PM	Project Milestone
PMR	Preliminary Monitoring Report
PCMS	Primary coolant make-up system (including let-down system T_K/T_B):
PN9	Temelín Road Map Project No. 9
PNAE-G	Russian Regulations
PORV	Power Operated Relief Valve
POSAR	Pre-Operational Safety Analysis Report
PRISE	Primary-to-Secondary
PRISE leak	PRImary to SEcondary Leak
PRT	Pressuriser Relief Tank
PRZ	Pressuriser
PRZ PORV	Pressurizer Power Operated Relief Valve

PRZ SL	Pressuriser Surge line
PRZ SV	Pressuriser Safety Valve
PSA	Probabilistic Safety Analysis
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review 10-years-cycle examination
PSV	Pressurizer Safety Valve
PSV43b	Pressuriser Safety Valve LOCA
p-T curves	Pressure versus Temperature plots
PTS	Pressurized Thermal Shock (rascher Abkühlvorgang bei hohem Druck)
PTS	Pressurized Thermal Shock
PTSA	Pressurized Thermal Shock Analysis
PWR	Pressurised Water Reactor
QA	Quality Assurance
RCC-M	Règles de Conception et de Construction des Matériels Mechaniques
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RED	Radiation Enhanced Diffusion
RELAP5	TH code (US NRC INEEL maintained)
Řež	NRI Řež Nuclear Research Center CZ
RMS	Root mean square
RNO	Return to Normal Operation
RPV	Reaktor pressure vessel
RPVI	Reactor Pressure Vessel Integrity
RSEM	French Code
RVLIS	Reactor Vessel Level Indication system
SA	Severe Accident
SAG	Severe Accident Guidelines
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SB32	SBLOCA with break diameter 32
SBLOCA	Small Break Loss of Coolant Accident
SCG	Severe Challenge Guidelines
SCRAM	Emergency shutdown of a nuclear reactor
SCST	Severe Challenge Status Tree
SDA	Steam Dump to Atmosphere
SDC	Steam Dump to Condenser
SEI	State Energy Inspectorate
SG	Steam Generator
SGSV	Steam Generator Safety Valve

SGT	Steam Generator tubes
SIR	Specific Information Request
SK-187	Manipulator for ISI
SKIN	Special Manipulator used by ETE
ŠKODA	Czech Company
SLB	Steam Line Break
SLB1B	Main steam line break by steam generator 1
SLB1B	Steam Line Break 1B
SLIC-Probes	Probes for Non-Destructive Testing
SMF	Stable matrix features
SMILE	Structural Margin Improvements in Aged-Embrittled RPV with Load History Effects
SN	Czech abbreviation for weld number
SONS	Czech Nuclear Regulatory Authority
STUK	Säteilyturvakeskus – Radiation and Nuclear Safety Authority, Finland
SÚJB	Czech Nuclear Regulatory Authority
SV	Safety Valve
TACIS	Technical Assistance to the Commonwealth of Independent States
TH	Thermal-Hydraulics
TK	Primary coolant make-up system (i.g. including lubrication)
TK/TB	Primary make-up/let-down system
TLFW	Total Loss of Feed Water
TOFD	Time-of-Flight Diffraction (UT)
TQ12, 22D01	Low Pressure Injection Pump
TQ13, 23D01	High Pressure Injection Pump
TQ2 system	Low Pressure Injection System (LPIS)
TQ3 system	High Pressure Injection System (HPIS)
TQ4 system	High Head Injection System (HHIS)
TSC	Technical Support Centre
TWE	CZ Abbreviation for Maximum Allowable Defect Height
UJE	Operator of NPP Temelín
UK Reg. 1994	United Kingdom Regulation 1994
UMD	Unstable Matrix Defect
UP	Upper Plenum
US	United States of America
USNRC	United States Nuclear Regulatory Commission
USSR	Union of the Socialist Sovjet Republics
UT	Ultrasonic Testing

VERLIFE	Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs
VLI	Verifiable Line Item: Monitoring topic for ETE
WCL	Weld centre line
WOG	Westinghouse Owners Group
WPS	Warm Pre-Stressing (effect)
WWER	Vodo-Vodyannoy Energeticheskiy Reactor – water-cooled, water-moderated, reactor; Soviet-design pressurised water reactor
YT11, 12, 13, 14 B01	Hydro accumulators
ZG3420	Appendix of the French Code RCC-M
γ-heating	Heating due to Gamma-Irradiation

FORMULA ENTRIES

	Fracture toughness-temperature curves for
$K_{Jc}[T]$	Static loading conditions
$K_{Id}[T]$	Dynamic loading conditions
$K_{Ia}[T]$	Crack arrest loading conditions
ΔRT_{PTS}	Maximum Allowable Shift of the Reference Transition Temperature (Screening Criterion of the U.S, Regulations)
$\Delta T_{41\ J}$	Temperature shift at the 41 J level
ΔT_{RTNDT}	Shift of the nil-ductility transition reference temperature
$1{,}1K_{IP} + K_{IS}$	Safety factor fort the primary and secondary stresses (Russian Code)
a, c	Minor resp. major ellipses half-diameter of the postulated defect
$^{a}/_{c}$, $2^{a}/_{c}$	Postulated semi-elliptical resp. elliptical defect aspect ratio
A_F	Embrittlement coefficient
$A_F^{290°C}$	Embrittlement coefficient at irradiation temperature 290 [°C]
A_F^T	Irradiation embrittlement factor at irradiation temperature T
c_p	Specific heat
Cu	Copper
C_v (T)	Charpy impact energy as function of temperature
DBTT	Ductile-to-brittle Transition Temperature
dK_I/dt	Temporal change of the stress intensity factor
E	Young's modulus
F_n	Neutron fluence
ND850	Nominal diameter 850 [mm]
K_I	Stress intensity factor
K_{Ic}	Fracture toughness
$K_{Ic(95\%)}$	Master curve 95%-percentile
$K_{Ic(05\%)}$	Master curve 5%-percentile
$K_{Ic(med)}$	Master curve (average)
$K_{Ic}(T)$	Temperature dependence of the fracture toughness
Mn	Manganese
MW	Megawatt
MW_t	Megawatt Thermal
n_a	Safety factor concerning crack size
Ni	Nickel
n_k	Safety factor on stress intensity
P	Pressure
P+0,2Cu	Phosphorus content +0,2 copper content (in %wt)
RT_{NDT}	Nil-ductility transition reference temperature (reference temperature for the ductile-brittle transition)

RT_{PTS}	Maximum allowable RT_{NDT} (Screening Criterion)
s	RPV wall thickness
S_N+S	RPV wall thickness + cladding thickness
T	Temperature
T_{47J}	Temperature at the 47 [J] level (Charpy curve)
T_{68J}	Temperature at the 68 [J] level (Charpy curve)
T_{70J}	Temperature at the 70 [J] level (Charpy curve)
T_{CV}	Temperature of Charpy tests were at least 0,89 [mm] lateral expansion or 68 [J] absorbed energy are measured
T_{irr}	Irradiation temperature
T_{IS}	Temperature of the inner RPV surface
T_k	Critical brittle fracture temperature
T_{K0}	Initial value of critical brittle fracture temperature
T_k^a	Maximum allowable critical brittle fracture temperature
T_o	Reference temperature indexed at distinct reference fracture toughness (e.g. 100 [MPa.m$^{1/2}$])
T_R	Reference temperature
σ_{om}	Stress residual maximum
σ_y	Yield stress
Φ	Neutron flux defined as the number of neutrons crossing a unit area per unit time in [neutrons/m^2.s]
$\Phi_{t>0,5}$	Neutron flux standard unit of neutron exposure with energy > 0,5 MeV in WWERs is about 1.1017 ^1_0n /m².s
$\Phi_{therm>1MeV}$	
$\Phi_{TI>0,5}$	Neutron fluence is the flux integrated over irradiation time for thermal neutrons with an energy > 0,5 MeV at 255 °C irradiation temperature for WWERs in [neutrons/m^2] End-of-life fluence for WWERs is about 7.10^{23} ^1_0n /m^2 ($\Phi T_{I>1,0}$ in PWRs about 1÷3 10^{23} ^1_0n /m^2 in BWRs about an order of magnitude lower).
ΔT	Safety margin for the critical temperature of brittleness
ΔT_F	Shift in T_K due to irradiation
ΔT_{Fr}	Residual shift in T_K after annealing
ΔT_n	Shift in T_K due to fatigue damage
ΔT_T	Shift in T_K due to thermal ageing
α	coefficient of linear thermal expansion
λ	Thermal conductivity
ν	Poisson's ratio
ρ	density

SI-UNITS AND OTHER

% vol.	Percent fraction of volume (gas)
% wt.	Percent fraction of weight (solids, liquids); also denoted as %
°C	Celsius (degrees) (temperature): 0 [°C] = 273,6 [K]
°C/h	Celsius per hour (temperature change with time) 1 [K/h] ≡ 1 [°C/h]
$^1/_a$	Events per year (frequency of events per year (of reactor operation))
$1/m^2$	Unit/Events per square meter
${}^1_0n$	Neutron
${}^1_0n/cm^2$	Neutron dose: composed unit [number of neutrons/cm²]
${}^1_0n/m^2$	Neutron dose: composed unit [number of neutrons/m²]
a	Year (time)
Bar	Bar (pressure difference) 1 [Bar] = 10^5 [Pa] (in excess of environmental)
Bar_{abs}	Bar absolute (pressure absolute) 1 [Bar] = 10^5 [Pa]
CF	Chemistry factor
cm	Centimetre (length)
cm/h	Centimetres per hour (speed (here speed of ablation))
cm²	Centimetre square (area)
d	Day (time)
g	Gram (weight)
h	Hours (time)
inch	Inches length (US Standard system of Units)
J	Joule (work) unit [Nm]
K	Kelvin (degrees) temperature or temperature difference
K/h	Kelvin per hour (temperature change with time) 1 [°C/h] ≡ 1 [K/h]
K/MPa	Kelvin/Mega Pascal: composed unit
kg/s	Kilogram per second (mass-flow)
kJ/mol	Kilo-Joule/Mol (work per Mol – work for/from chemical reaction)
km	Kilometre (distance, length)
kPa	Kilo-Pascal
l/h	Volume flow: composed unit 0,277778 [cm³/s]
m	Meter (length)
m^2/MW_{th}	Specific surface for heat transfer
m³	Cubic meter (volume)
MeV	Mega Electron Volt
mm	Millimetre (length)
MPa	Mega-Pascal

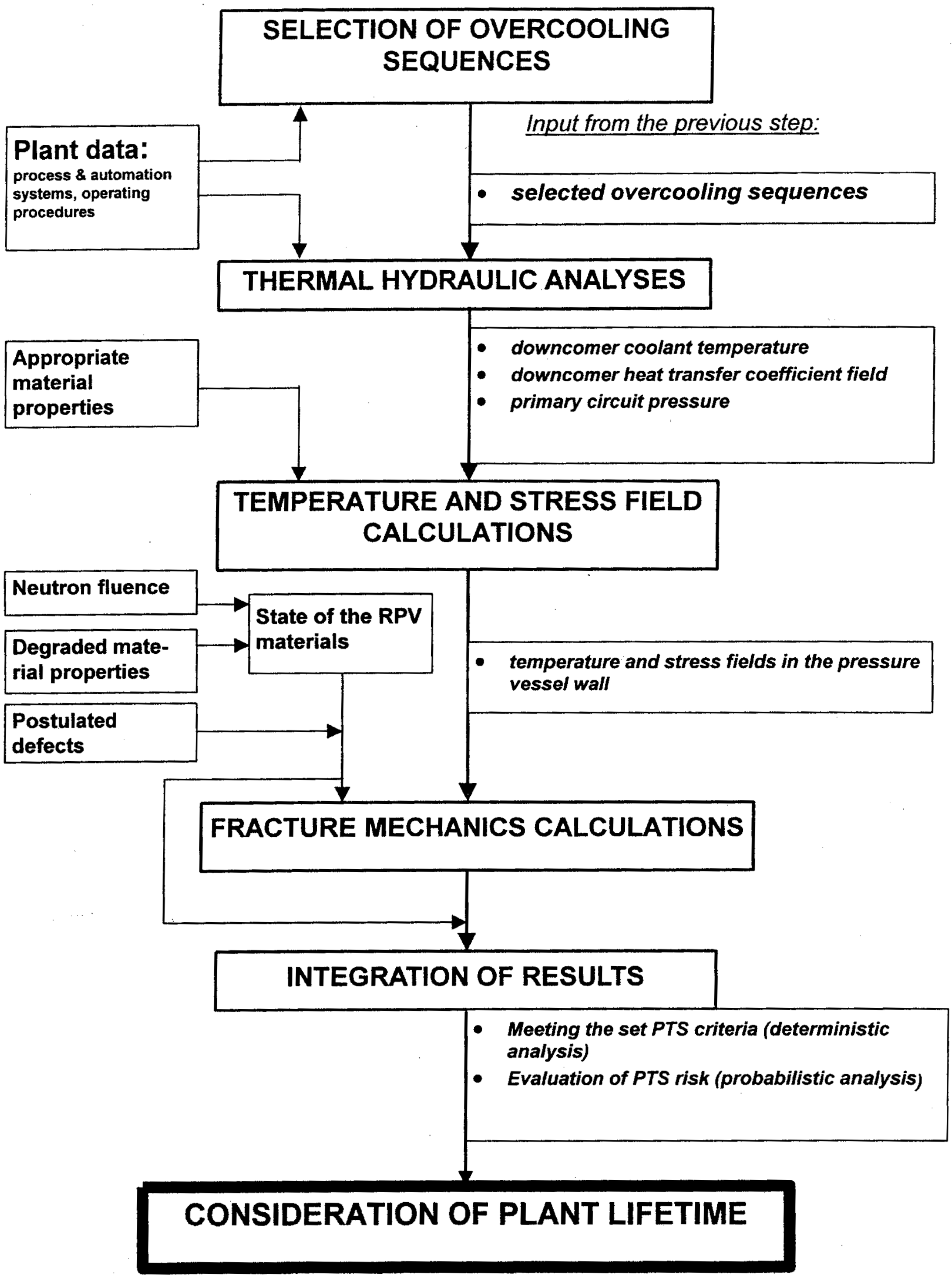
$MPa.m^{1/2}$	Mega-Pascal per square meter (Stress intensity factor)
MPa_{abs}	Mega-Pascal absolute (pressure)
MW_e	Electrical power output/demand
Pa	Pascal (pressure) 1 [Pa] ≡ 1 [N/m²]
s	Second (time)
Sv	Sievert (effective dose (received by humans from ionising radiation))
Sv/a	Sievert per year (of operation) radiation Risk to the public resulting for one year of operation
t	Ton (weight)
t/h	Tons per hour (mass flow)
1	Dimensionless units, relative units

ANNEX A

PTSA BASIC CONCEPT OF CONDUCT

Concept of Conduct of a PTSA

(this procedure was also applied for the Austrian Benchmark Exercise [ANNEX D])



ANNEX B

VERLIFE FOR ORIENTATION

EU Nuclear Fission and Radiation Protection Projects Selected for Funding 1999 – 2002, [VAN GOETHEM 2003]

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*[ed. rem.] The **VERLIFE** Program aims in the Nuclear Fission and Radiation Protection Research context:*

Harmonise VVER and PWR Codes and Procedures for plant life management and participate in the development of a common safety culture.

The FP-5 project IMPAM-VVER is addressing a safety relevant issue identified in recent studies on VVER safety. It investigates effective means and criteria for primary depressurisation during small loss of coolant accident (SBLOCA) including feed and bleed operation. The resulting knowledge will effectively contribute to the safety in all VVER countries. Two FP-5 concerted actions are dedicated in harmonising safety culture within an enlarged Europe.

The project VERSAFE brings together utilities from some of the Central and Eastern European Countries: common guidelines have been produced for the implementation of techniques in two areas, plant modernisation and severe accident management. VERLIFE is creating a “unified procedure for lifetime assessment of components and piping in VVER type nuclear power plants” based, in a first step, on former Soviet rules and codes. Later on, a critical analysis of possible application to some PWR type components will be done, with the aim to harmonise VVER and PWR Codes and Procedures.

[ed. rem.] VERLIFE Project Description and Participants as contained in: [EUR 20617]

Project Acronym: **VERLIFE**

UNIFIED PROCEDURE FOR LIFETIME ASSESSMENT OF COMPONENTS AND PIPING IN VVER NPPS

Project Description

Main goal of the project will be a preparation, evaluation and mutual agreement of a "Unified procedure for Lifetime Assessment of Components and Piping in VVER Type Nuclear Power Plants". This procedure should be based on former Soviet rules and codes, as VVER components were designed and manufactured in accordance with requirements of these codes and from prescribed materials. Then, critical analysis of possible application of some approaches used in PWR type components will be performed and such approaches will be incorporated into the prepared procedure as much as possible with the aim of a harmonisation of VVER and PWR Codes and procedures. Preparation of a Unified Procedure for VVERs operating in Finland, Czech Republic, Slovak Republic and Hungary will increase the level of lifetime/integrity evaluation.

Elaboration of a "Unified Procedure for Lifetime Assessment of Components and Piping in VVER NPPs" to be usable in nuclear power plants in Finland, Czech Republic, Slovak Republic and Hungary for Periodic Safety Reports and Components/Plant Life Management. Application of this Procedure will unify assessment methods in individual nuclear power plants and will harmonise approaches between VVER and PWR lifetime/evaluation assessments.

The Concerted Action will be based on the partner's meetings that are the main method of sharing the status of procedures for lifetime evaluation in individual countries and the partner's views of procedure unification.

The practical project will be divided into three principal parts – Work Packages:

- Analysis of approaches, methods and material properties applied in WER Codes and standards and PWR Codes with the aim of harmonisation of WER approaches with PWR ones as much as possible, preparation of a common status report of procedures used in individual countries
- Elaboration of a proposal of a "Unified Procedure for Lifetime Assessment of Components and Piping in VVER NPPs" based on VVER Codes and standards using agreed approaches from PWR Codes this proposal will be discussed in meetings of Task groups organised in accordance with specialities of individual chapters in Procedures
- Finalisation of the Procedure after mutual agreement of applied approaches and methods to be usable in all four VVER operating countries.

Participation in the Project will be held by experts from nuclear power plants technical support organisations and national regulatory bodies to prepare a Procedure that will be applicable for Periodic Safety Reports and Plant Life Management of individual nuclear power plants.

Lifetime assessment of individual components and piping in nuclear power plants (NPP) is a mandatory part of every Periodic Safety Report as well as it is necessary for component/ plant life management and potential plant life extension.

Today, no legal procedures or standard guidelines exist for lifetime/integrity assessment of components and piping in operating NPPs of VVER type. Former Soviet rules and standards were prepared and approved only for design and manufacturing stage of NPPs. These rules/ standards mostly are not applicable for operating plants or they need some modifications and extensions to be usable also for operating components. Approaches used in VVER Codes and standards are in some parts different than they are applied in PWR ones, thus a comparison of lifetime assessment using these two types of codes could be different and comparable.

Project Acronym:	**Project Reference:**	**FIKS-CT-2001-20198**	**Contract Type:**	**Coordination of research actions**
VERLIFE	Start Date:	2001-10-01	Duration:	24 months
	End Date:	2003-09-30		
	Organisation	Address	Type	Contact
Coordinator				
Nuclear Research Institute Řež A.S.	Division of Integrity and Technical Engineering	Řež 130250 68 Řež – PrahaCzech Republic	Research	Pazdera, F. (Dr)
Participants				
Bulgarian Academy Of Sciences	Institute of Metal Science	67 Shipchenski Prohod Street1574 Sofia Bulgaria	Research	Arsov
Škoda JS A.S.	Structural Safety Laboratory	Orlik 266316 06 Plzen Czech Republic	Other	Svitak, Fr.
Czech Energy Company, Inc.	Lifetime Management Department	Dukovany Nuclear Power Plant Station 675 50 Dukovany Czech Republic	Other	John, A.
Slovenske Elektrarne, A.S.	Department of Nuclear Safety and Crisis Management	Hranicna 1282736 Bratislava Slovakia	Other	Gunis, R.
Nuclear Regulatory Authority of the Slovak Republic	Division of Systems and Components	Okruzna 5 91864 Trnava Slovakia	Other	Lipar, M.
Institute Of Applied Mechanics Brno, Ltd.	Diagnostic Systems of Technological Plant Equipment	Veveri 95 611 39 Brno Czech Republic	Research	Vejvoda
Czech Energy Company, Inc.	Diagnostics and Ageing Management Department	Temelín Nuclear Power Plant Station 373 05 Temelín Czech Republic	Other	Hezoucky F.
Vuje Trnava Inc – Engineering, Design and Research Organisation	Division of Diagnostics of Nuclear Power Installations	Okruzna 5 918 64 Trnava Slovakia	Research	DUGOVIC, M.
Fortum Nuclear Services Oy	Structures and Quality	Rajatorpantie 8 00048 Espoo Finland	Other	Vaeyrynen, H.
State Office for Nuclear Safety		Senovazne Nam. 9 110 00 Praha 1 Czech Republic	Other	Boehm, K.
KFKI – Atomic Energy Research Institute – Hungaria	Fuel and Reactor Materials Department	Konkoly-Thege Miklos ut. 29-33 P.O.Box 49 1121 Budapest Hungary	Research	Gado, J. (Dr)

[BRUMOVSKÝ 2003]

UNIFIED PROCEDURE FOR LIFETIME ASSESSMENT OF COMPONENTS AND PIPING IN WWER NPPS (VERLIFE)

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[1)] Nuclear Research Institute Řež (CZ), [2)] State Office for Nuclear Safety (CZ) [3)] Dukovany Nuclear Power Plant (CZ) [4)] ŠKODA JS plc (CZ) [5)] Fortum Nuclear Services Oy (FI) [6)] Slovenské elektrarne (SR) [7)] VUJE Trnava (SR) [8)] KFKI-AEKI(HU) [9)] Nuclear Regulatory Office (SR) [10)] Temelín Nuclear Power Plant (CZ) [11)] Institute of Applied Mechanics (CZ) [12)] Institute of Metal Science (BG)

SUMMARY

Activities of the project have been concentrated in a preparation, evaluation and mutual agreement of a “Unified procedure for Lifetime Assessment of Components and Piping in WWER Type Nuclear Power Plants” that was the planned goal of the project. This procedure is based on former Soviet rules and codes, as WWER components were designed and manufactured in accordance with requirements of these codes and from prescribed materials. Then, critical analysis of possible application of some approaches used in PWR type components was performed and such approaches have been incorporated into the prepared procedure as much as possible with the aim of a harmonisation of WWER and PWR Codes and procedures.

Preparation of a Unified Procedure for WWERs operating in Finland, Czech Republic, Slovak Republic, Hungary and Bulgaria will increase the level of lifetime/integrity evaluation in these countries and will help to elaborate a unified approach and fully comparable results between individual plants and countries.

Then, harmonisation with PWR codes allows obtaining results that will be comparable, reliable and more sophisticated as similar approaches will be used in both types of reactors. The project was performed within the 5th Framework Programme; it was planned and performed in 24 months, the project was finished on September 30, 2003.

A. INTRODUCTION

Lifetime assessment of individual components and piping in nuclear power plants (NPP) is a mandatory part of every Periodic Safety Report as well as it is necessary for component/ plant life management and potential plant life extension. In the same time, such assessment is also necessary for safe operation of components in NPPs. Today, no legal procedures or standard guidelines exist for lifetime/integrity assessment of components and piping in operating NPPs of WWER type. Former Soviet rules and standards had been prepared and approved only for design and manufacturing stage of NPPs. These rules/standards mostly are not applicable for operating plants or they need some modifications and extensions to be usable also for operating components. Approaches used in WWER Codes and standards are in some parts different than they are applied in PWR ones, thus a comparison of lifetime assessment using these two types of codes could be different and non-comparable.

Additional goal of this Concerted Action has been in creating a network of the safety managers and experts of the plants together with experts from Technical Support Organisations and from national regulatory bodies that are foreseen to operate WWER type reactors within the European Union during the first decades of this century.

B. WORK PROGRAMME

The Concerted Action was based on the partners´ meetings that were the main method of sharing the status of procedures for lifetime assessment of individual components and piping in WWER NPPs in members´ countries.

The Consortium has been co-ordinated in such a way to:

- Group experts from technical support organisations that are incorporated in WWER component lifetime assessment in Finland, Czech Republic, Slovak Republic and Hungary – FORTUM Nuclear Services Ltd. in Finland, Nuclear Research Institute Řež plc and Institute of Applied Mechanics in Czech Republic, VÚJE Trnava a.s. in Slovak Republic and AEKI Atomic Energy Research Centre in Hungary, and Institute of Metal Science in Bulgaria,
- Include experts from nuclear regulatory bodies that are connected with evaluation of such assessments and/or their acceptance – State Office of Nuclear Safety of Czech Republic, Nuclear Regulatory Office of Slovak Republic,
- Include specialists from nuclear power plants that are responsible for component lifetime assessment and/or plant life management – FORTUM Nuclear Services Ltd. for Loviisa NPP in Finland, ČEZ a.s. for NPP Dukovany and for Temelín NPP in Czech Republic, Slovenske elektrárne a.s. for NPP Jaslovské Bohunice and Mochovce in Slovak Republic, and NPP Paks through AEKI,
- Use experience from components design, stress analysis, lifetime evaluation and manufacturing experience – ŠKODA JS a.s. in Czech Republic (main manufacturer for WWER components for Czech Republic, Slovak Republic and Hungary, resp. also former Germany Democratic Republic, Poland and Bulgaria).

The practical work was carried out, in principal, in meetings of Task groups.

A common kick-off meeting of all partners started the work and review the current status. In the kick-off meeting the structure of the “Unified Procedure” and expected results were proposed, discussed and agreed. For evaluation of the “Unified Procedure”, Task Groups of experts have been organised in the following main directions that cover all main scientific fields of the “Procedure…”:

- Fracture – application of fracture mechanics to integrity and lifetime, defects allowance,
- Corrosion-mechanical damage – corrosion problems in integrity and lifetime assessment,
- Fatigue – mechanical and thermal fatigue evaluation based on design and real operating regimes,
- Material ageing – definition, material testing and damage evaluation,
- Reactor dosimetry – determination of neutron fluences by calculations and measurements.

In order to ensure that Task Groups work proceeds according to the common aim of the participating organisations, a midterm review meeting was scheduled. The objective of the meeting was to:

- discuss eventual needs to re-orient the Task Group work, and
- discuss eventual reorganising the Concerted Action by inviting new members.

As a result, extension of the project by three new organisations from Czech Republic and Bulgaria was successfully agreed.

Four meetings of all Task Groups were carried out, they have completed their work in discussion and agreement on text of individual parts of the Procedure. Their results are now collected in the Final draft of the “Unified Procedure” as it was agreed in the final project meeting in the beginning of September 2003. Thus, the “Unified Procedure” is taken as accepted by all participants and will be proposed to individual national Nuclear Regulatory Authorities for their acceptance and approval as a basis document for lifetime evaluation and preparation of Periodic Safety Reports.

Workshop on the scope and approach of the Unified Procedure was prepared and organised at the end of September 2003 for end-users of the Procedure – plant owners and operators, technical support organisations and regulatory bodies in partners´ countries. This Workshop was held for organisations from the Czech Republic, Slovak Republic, Hungary, Finland and Bulgaria.

C. MAIN ACHIEVEMENTS

In preparation of the "Unified Procedure" the following principles and inputs have been agreed:

- WWER components were designed and manufactured in accordance with former Soviet rules and standards [1],
- IAEA activities in the field of WWER components integrity assessment [2],
- Approaches applied in PWR components integrity and lifetime evaluation,
- Last developments in fracture mechanics and their application to component integrity.

Large effort was concentrated on creation and critical analysis of material databases of main WWER component materials – fracture toughness, crack growth rate, corrosion resistance, radiation damage. On the bases of these databases, necessary design curves for individual material properties have been proposed and put into the "Unified Procedure".

Main difference between original Soviet rules [1] and the "Unified Procedure" can be found in strict application of "Master Curve" approach for component integrity assessment as real material properties are used without necessity of any empirical correlations between different type of test results (traditional transition temperatures based on Charpy impact test data are allowed as a secondary alternative).

Such advantage can be seen from comparison of data for e.g. base metal of 15Kh2MFA for reactor pressure vessel of WWER-440 units – figure 1 shows large scatter of fracture toughness data based on "critical temperature of brittleness, T_k" (based on Charpy impact test data) while figure 2 shows the same data but correlated directly with "reference temperature, T_0" using "Master Curve" approach. More than 1,200 data from different WWER countries were collected.

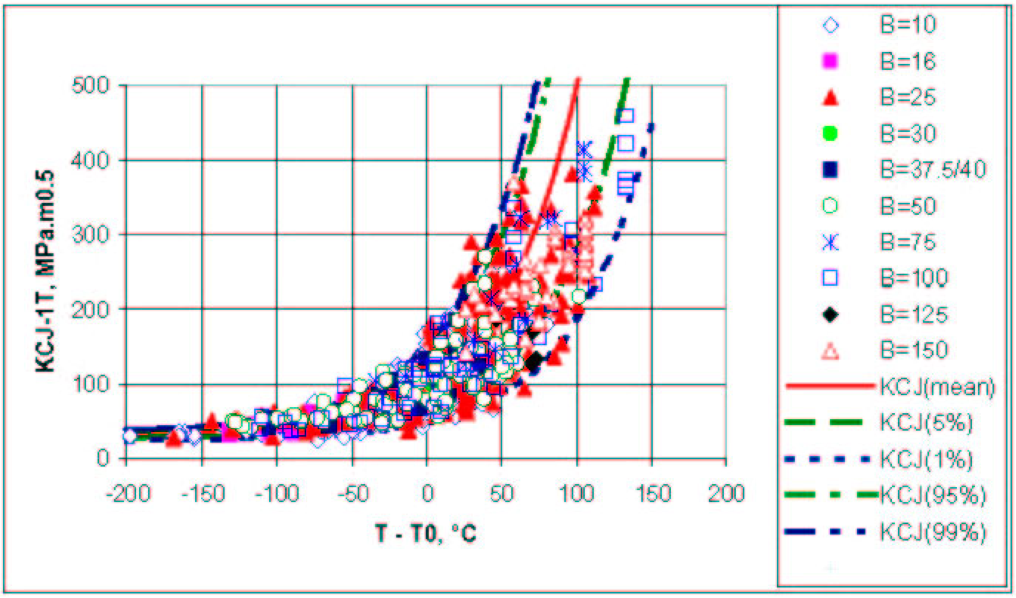

Figure 1: Temperature dependence of static fracture toughness data of 15Kh2MFA type steel (base and weld metals) for WWER-440 reactor pressure vessel correlated with transition temperature T_{k0} (B = specimen thickness, $[K_{IC}]^3$ = generic design fracture toughness curve)

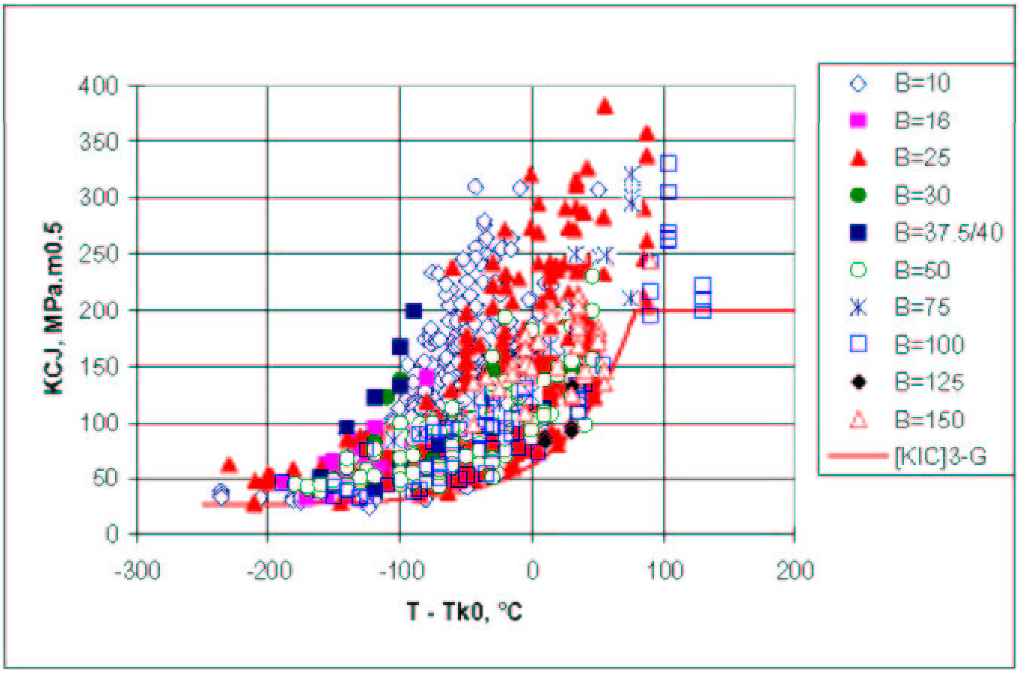

Figure 2: Temperature dependence of static fracture toughness data adjusted to 1 [inch] thickness of 15Kh2MFA type steel (base and weld metals) for WWER-440 reactor pressure vessel correlated with reference temperature T_0 in accordance with "Master Curve" approach (B = specimen thickness)

The structure of the "Unified Procedure" covers all-important parts of lifetime and integrity assessment, as they are required for Periodic Safety Reviews and plant life management programmes:

1. Scope of the procedure
2. General requirements, definitions and abbreviations
3. General requirements to lifetime assessment
4. Assessment of resistance against non-ductile failure for normal, upset, emergency operating conditions and for pressure tests
5. Assessment of fatigue damage
6. Assessment of corrosion-mechanical damage
7. Evaluation of allowance of defects found during in-service inspection
8. Final lifetime assessment

Detailed procedures or requirements are summarized in the following appendices:

I. Structure of the Report Assessing Residual Lifetime of the Equipment
II. Procedure for Determination of Radiation Field in Reactor Pressure Vessel Including Monitoring
III. Assessment of Degradation of Properties of RPV Materials
IV. Determination of Values of Stress Intensity Factor KI
V. Determination of Reference/Design Fracture Toughness Curve Including "Master Curve" Approach
VI. Requirements for Pressurized Thermal Shock (PTS) – Selection and Thermal Hydraulic Calculations
VII. Residual Lifetime of The Equipment Damaged by Fatigue due to Operating Loading
VIII. General Recommendation for Piping and Components Temperature Measurement
IX. Assessment of Corrosion-Mechanical Damage in Materials
X. Schematisation of Flaws
XI. Tables of Allowable Sizes of Indications Found During In-Service Inspections
XII. Evaluation of Defect Allowance in Components
XIII. Assessment of Acceptability of Flaws in Austenitic Piping
XIV. Computational Assessment of Allowability of Flaws in Carbon Steel Piping
XV. Material Properties to Be Used for Temperature and Stress Fields Calculations

Within the Assessment of Reactor Pressure Vessel Resistance against Fast Fracture in the "Unified Procedure…", the following principal changes in comparison with original rules [1] have been implemented:

- size of the postulated defect for fast fracture evaluation as well as for fatigue and corrosion-mechanical damage is defined in correlation with in-service inspection methods and qualification,
- method for evaluation of allowance of defects found during in-service inspections is given,
- allowable sizes of defects found during in-service inspections are calculated on the bases of fracture mechanics and material properties,
- method for transformation of indications found during in-service inspections into calculated defects is described,
- procedure for evaluation of surveillance specimens test data for their use in integrity assessment is given,

- method for evaluation of corrosion-mechanical damage in some specific components is described,
- material properties (crack growth rates) in primary water environment are summarized,
- unified material properties for temperature and stress fields of reactor pressure vessels are summarized.

This "Unified Procedure" has been prepared for pressurized components of primary circuit of WWER-440 and WWER-1000 units, even though it shall be used for safety related components of other circuits, too.

D. DISSEMINATION AND EXPLOITATION OF THE RESULTS

This "Unified Procedure" has been prepared in close co-operation between operators, technical support organizations and national regulatory authorities from several WWER countries. Regarding the necessity of such document for preparation of Periodic Safety Reviews as well as Reports for License Renewal in most of countries from the project, this the final version of the document will be translated into national languages and supplied to individual national regulatory authorities for their approval. As some of them were active in the document preparation, it is expected that they will accept the document without any delay and principle changes.

E. CONCLUSIONS

The Concerted Action for the project VERLIFE has been successfully finished in accordance with the plan and the contract. The "Unified Procedure for Lifetime Assessment of Components and Piping of WWER NPPs" has been prepared, discussed, evaluated and finally agreed and accepted by all participants. This "Procedure" represents a procedure for WWER components based on former Russian codes and rules but harmonised with approached used in PWR type plants. Thus, this "Procedure" is now in good agreement with the state-of-the-art of the knowledge in the field. The "Procedure" will be now proposed to national Nuclear Regulatory Authorities for approval and use.

During the project performance, a very effective and qualified group of experts have been created – it would be against common sense not to continue in these activities also in the future as all standards/rules must be living documents that need their revision every two/three years and without such revisions they will be dead and non-applicable any more in a near future.

REFERENCES

[1] Standard for Strength Calculations of Components and Piping in NPPs, PNAE G-7-002-86, Energoatomizdat, Moscow, 1989

[2] Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants, IAEA-EBP-WWER-08, Vienna, 1997

ANNEX C

LIST OF AUSTRIAN PROJECTS

Austrian Projects Identification

PN 1	Severe Accidents Related Issues	[Item No. 7a]*
PN 2	High Energy Pipe Lines at the 28,8 m Level (AQG/WPNS country specific recommendation)	[Item No. 1]*
PN 3	Qualification of Valves (AQG/WPNS country specific recommendation)	[Item No. 2]*
PN 4	Qualification of Safety Classified Components	[Item No. 5] *
PN 5	Chapter V – Environmental Impact Assessment	
PN 6	Site Seismicity	[Item No. 6]*
PN 7	Severe Accidents Related Issues	[Item No. 7b]*
PN 8	Seismic Design	
PN 9	Reactor Pressure Vessel Integrity and Pressurised Thermal Shock	[Item No. 3]*
PN 10	Integrity of Primary Loop Components – Non Destructive Testing (NDT)	[Item No. 4]*

** The Items are related to ANNEX I of the "Conclusions of the Melk Process and Follow-up"*

ANNEX D

AUSTRIAN BENCHMARK EXERCISE

WWER-1000/320: List of PTS Initiating Events

At present guidelines for plant-designers recommend the following list of initiating events (IEs) to be considered for WWER-1000 PTS analyses. These IEs were used to explore the PTS in relation to the WWER-1000 emergency operation procedures and the thermal load transients as indicated in the second column:

IAEA-EBP-WWER-08, IAEA, Vienna, April 1997		Austrian Benchmarks
#	**Candidate Transient**	**IRR/ARCS treatment**
1.	Spectrum of postulated piping break within the reactor coolant pressure boundary.	performed DEGB, limited LOCA intermediate
2.	Rupture of the line connecting the pressurizer and a pressurizer safety valve.	prepared
3.	Inadvertent opening of one pressurizer safety valve.	Performed considered with 8.
4.	Leaks from the primary to the secondary side of the steam generator: • SG tube rupture • Primary collector leaks up to cover lift-up.	omitted
5.	Inadvertent opening of one check or isolation valve separating reactor coolant boundary and low-pressure part of the system.	omitted
6.	Inadvertent actuation of ECCS during power operation.	considered with 3.
7.	Chemical and volume control system malfunction that increases reactor coolant inventory.	omitted
8.	Inadvertent opening of one steam generator safety or relief valve or turbine bypass valve.	considered with 9.
9.	Spectrum of steam system piping break inside *(rem. IC)* and outside of containment *(rem. OC)*	performed for *OC* considered with 8.
10.	Feed-water piping break.	omitted

(List according to: Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants, IAEA-EBP-WWER-08, IAEA, Vienna, April 1997)

In the third column the transients as selected for the Austrian benchmark exercise are denoted addressing WWER-1000 PTS events consequences to the RPV wall. Most of the work accomplished for a generic WWER set-up compares well to the actual ETE situation.

ANNEX E

EXTRACT FROM THE "SPECIFIC INFORMATION REQUEST BY THE AUSTRIAN EXPERTS' TEAM"

Technical Support
for the monitoring on the technical level of the implementation of
Annex I and Annex II of the
Conclusions of the Melk process and follow-up

Project No. 9

Reactor Pressure Vessel Integrity and Pressurised Thermal Shock

PM2

Specific Information Request

DOCUMENT NO: **IRR/ARCS**-520-PN9-02-08

TITLE: Temelín Road map, PN9 – PM2: Specific Information Request

TABLE OF CONTENTS

ABBREVIATIONS LIST

AM	Accident Management
ČEZ, a.s.	Check utility – joint stock company **ČEZ**, a.s.
EGP	The Czech Main Designer for Temelín NPP
EOP	Emergency Operation Procedures
ESFAS	Emergency Safety Features Actuation System
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
ISI	In-Service Inspection
LBB	Leak-Before-Break
MSIV	Main Steam Isolation Valves
NDT	Non-Destructive Testing
NPP	Nuclear Power Plant
PORV	Power Operated Relief Valve
POSAR	Pre-Operational Safety Analysis Report
PSR	Periodic Safety Review
PTS	Pressurised Thermal Shock
RCS	Reactor Coolant System
RPVI	Reactor Pressure Vessel Integrity
SAMG	Severe Accident Management Guidelines
SG	Steam Generator
SSC	Structures, Systems and Components
SÚJB	Czech Nuclear Regulatory Authority
V&H	Vertical and Horizontal (Evaluation), refers to specific segments of the PN9 project
VLI	Verifiable Line Item

1. INTRODUCTION

The report presents the information entries related to Reactor Pressure Vessel Integrity and PTS analyses which would support in obtaining profound answers to the VLIs for ETE, help the preparation of the Austrian Delegation for the discussion to take place during the Workshop on Reactor Pressure Vessel integrity and PTS analyses issues (planned for the first half of 2004 according to the Roadmap). This includes information on the plant specific analyses used in the development of PTS related measures, neutron embrittlement mitigation strategies considered, performance of the RPV and its verification based on administrative arrangements and procedures related to the development and implementation of PTS Guidelines, and plant specific design features relating to PTS phenomenology and RPV integrity preserving strategies. All available information addressed in the SIR report will also be used to provide input for an independent analysis of selected PTS scenario(s), planned to be conducted before the workshop.

With consideration of the categories indicated at the VLI and the technical issues, which are relevant to the evaluation of the plant status in the area of RPV Integrity and PTS the following areas of topics were compiled:

Regulatory Requirements and Applicable Standards

RPV-Related Plant Programs

General Design Considerations, Design Verification and Hardware Performance

Detailed Information on the Selected RPV Design Features and PTS Management: – Material Properties and NDE Aspects

Detailed Information on the Selected RPV Design Features and PTS Management – Thermal-Hydraulics Aspects

PTS Related Administrative Arrangements and Procedures

PTS Technical Guidelines, Procedures or Instructions

Embrittlement Related Records, Reports and Current Status of Surveillance Programmes

Evaluations by Third Parties

Chapter 2 contains a detailed listing structured according to the nine topics enumerated above, Chapter 3 specific references of a more general nature identified in preparing this SIR and Chapter 4 lists additional references of various sources.

Information already provided during the course of the activities carried out under the Melk these documents are available in or have been accessible to IRR Protocol in 2000-2001, and additional information obtained independently includes[66]:

[1] F. Pazdera, I. Váša & J. Žd'arek, VVER operational safety improvements: lessons learned from European co-operation and future research needs, Nuclear Engineering and Design 221 (2003), 193-204.

[2] Information provided by the SÚJB to the Austrian government on 2 September, 2000. SÚJB,

[3] Temelín Preoperational Safety Analysis Report, Revision 1, December 1999 (reviewed at the Temelín NPP December 2000 to January 2001).

[4] Kujal & J. Duspiva (Nuclear Analysis for the WWER-1000 Unit 1, presented at Severe Accident and Risk Management 1997 (SARM '97).

[66] these documents are available in or have been accessible to IRR.

[5] V. Pistora, Methodology of the structural part of the PTS assessment for the NPP Temelín, Workshop at SÚJB, Prague, Czech Republic, 7-8.November, 2002

[6] Temelín Probabilistic Safety Assessment, 1995 (reviewed at the Temelín NPP December 2000-January 2001).

[7] Minutes of the Trilateral Meeting February 27, 2001 concerning Issue No. 9: Reactor Pressure Vessel Embrittlement and Pressurized Thermal Shock

[8] Navody a doporuceni pro hodnoceni zivotnosti tlakove nadoby a vnitrnich casti reaktoru je VVER behem provozu JE, Prag, 1998

[9] Normy rasceta na procnost oborudovania i turboprovodov atomnych energeticeskich ustanovok, Energoatomizdat, 1989

[10] M. Brumovský, J. Brynda, Temelín RPVs Neutron Embrittlement, Nuclear Research Institute Řež plc, ŠKODA Jaderne Strojirenstvi a.s. a.s., Trilateral Technical Meeting, Řež, CZ, 26-27 February, 2001

[11] Rules for Design And Safe Operation of NPP Components, Research And Test Nuclear Reactors And Equipments (1973)

[12] Rules for Design And Safe Operation of Components and Piping of NPPs (PNAE G-7-008-89)

[13] Standards for Strength Calculation of Reactor Components, Steam Generators, Vessels and Piping of Nuclear Power Plants, Test And Research Reactors And Appliances (1973)

[14] Standards for Strength Calculation of Equipment and Piping of Nuclear Power Plants (1989)

[15] Base Regulations for Welding And Cladding of Components In NPPs, Test and Research Reactors And Equipments (OP 1513-72) (1972)

[16] Regulations for Inspections of Welded Joints In Nuclear Power Plants, Experimental and Test Reactors (PK 1514-72) (1972)

[17] Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants (IAEA-EBP-WWER-08, 1997)

[18] Requirements for Lifetime Evaluation of WWER Reactor Pressure Vessels and Internals During Their Operation (Sons, 1998)

[19] ASI (Czech Association of Mechanical Engineers) Codes for Reactor Components

Remark:

The documents cited under [11] to [18] have been declared by the Czech side to govern the licensing requirements. They are therefore the basis for any monitoring approach.

2. LIST OF DOCUMENTS

The Specific Information Request (SIR) has been prepared by joint effort of all Partners contributing to the project. The list provided below indicates the connection with the specific segments of the project ('horizontal' and 'vertical' evaluations) and the interrelation to VLIs of the SIR categories (H for 'horizontal' and V for 'vertical' evaluation segment).

I REGULATORY REQUIREMENTS AND APPLICABLE STANDARDS (VLI I)

1. Relevant Czech National regulations as well as decisions, orders, instructions or communications coming from the regulatory body (SÚJB) on RPV integrity and PTSA at Temelín NPP (H&V).
2. A list of full titles of international codes and/or national standards used/ referred to or required by the regulatory body or Temelín NPP in the planning, performance and evaluation of the PTS events and RPV integrity assurance (H&V).

II RPV-RELATED PLANT PROGRAMS (VLI II + III)

1. Plant program for the implementation of PTS related and RPV integrity assurance devoted measures at the Temelín NPP (H).
2. Plant program and schedule for the implementation of PTS related effort and for RPV surveillance program planning, performance, and evaluation, including staff training involved in PTS handling and RPV integrity verification (H).

III GENERAL DESIGN CONSIDERATIONS, DESIGN VERIFICATION AND HARDWARE PERFORMANCE (VLI III)

1. Chapter of the POSAR of Temelín NPP on PTS and PTSA, and on RPV integrity (H).
2. Specification of PTS scenarios to be investigated, in order to support development of both PTS Mitigation and Avoidance Programs at Temelín NPP (including modification of systems, operator actions or lack of actions, computer codes used, organization which performed the calculation, etc.) and an overview of results obtained from these analyses (H).
3. Information on computer codes used for PTSA (including information on development of VVER-specific models developed for the VVER 1000 and the use of codes oriented towards e.g. specific embrittlement consequences phenomena, heat transfer codes applied, boundary layer considerations etc. (H).
4. List of documents related to design verification made ex posteriori for PTS – in particular the following (V&H):
5. data, used models and obtained results concerning thermal hydraulic MIXING calculations as performed for ETE.
6. data representing the residual stress distribution due to the presence of cladding as assumed for the RPV
7. the stress-free temperature selected
8. list of high ranking codes, standards regulations and guidelines respected for design verification (in particular with regard to i to iii above)
9. specification for neutron fluence determination and verification

10. specification for establishing embrittlement prediction and verification
11. List of PTS strategies considered for Temelín NPP and a related overview (H).
12. Sample documentation/report that illustrates the presentation of PTS strategies (H).
13. List of plant systems/equipment considered in the PTS strategies and information on operability of this equipment under PTS conditions (cross references to specific AM strategies, where applicable) (H&V).
14. List of monitoring/instrumentation systems/equipment considered in PTS strategies and information on operation of this hardware under adverse operation conditions (cross references to specific AM strategies, where applicable) (H&V).
15. The range of instrumentation that can be used for PTS purposes, namely for measuring water level on SG secondary side, water level in the RPV, core coolant temperature, primary coolant temperature, radiation level in containment, water level in containment, pressure in containment, and hydrogen concentration in containment rooms (H&V).
16. List of system/equipment enhancements introduced or planned for PTS and RPVI (H).
17. List of hardware enhancements introduced or planned in relation to PTS and RPVI (H).
18. List of accident scenarios considered for PTS consequences (using plant simulator or any other simulation tool) (H).
19. Document(s) describing screening criteria and their application for selecting accident scenarios and accident phenomena for inclusion in the process of PTSA for the Temelín NPP (V).
20. Identification of the result of the screening and the subsequent PTSA efforts to eliminate vulnerabilities to PTS (IAEA PTS guidance terminology) for Temelín NPP and the corresponding EOPs management strategies which address these vulnerabilities (V).
21. Description of the selected surveillance program for analysis of the RPV embrittlement, time schedule of the planned sample withdrawals, and results of already tested surveillance samples (H&V).
22. Feasibility studies, owner's specifications, and decision documents concerning PTS mitigation measures considered but not implemented at Temelín, such as low leakage core, absorbing material distribution, core reshuffling strategies and preheating of the emergency coolant, coolant distribution sparger installation, efficiency verification for extremely fast depressurization, and any other measure demonstrated feasible to avoid or mitigate PTS consequences to the RPV integrity (H&V).
23. Plans for validation and verification of RPVI insofar as material degradation is concerned (including ageing, operational crack growth, and transient characterization etc.) (H).

IV DETAILED INFORMATION ON THE SELECTED RPV DESIGN FEATURES AND PTS MANAGEMENT – MATERIAL PROPERTIES AND NDE ASPECTS

1. Design requirements and capabilities of the Reactor Pressure Vessel as described in the specifications, approved and required by the licensing authority (H&V).
2. Design data include all PTS and RPVI precautions taken or added as well as the means for verification of the actual status of the RPV, thresholds of the relevant parameters (critical crack size, critical temperature, defect detectability, ...) (H&V).
3. Design parameters and geometry data (diameters, flow areas and elevations) of RPV and internals, primary loop and secondary pipelines until MSIVs, including relevant drawings, material properties (chemical composition of the base material and the welds, mechanical properties including specified critical values, embrittlement characteristics). The material properties request applies also to the surveillance program samples. (H&V).

4. Documents on material properties (chemical composition of the base material and the welds, mechanical properties including specified critical values, embrittlement characteristics). The material properties request applies also to the surveillance program samples (H&V).
5. Parameters relating to emergency coolant water availability change over time during an accident (H).
6. Mapping of the original deficiencies (flaws, cracks, inclusions, etc.) detected within a qualified NDT program after manufacturing, demonstration of their coverage during ISI, detection repeatability etc. (H).
7. Mapping of the original deficiencies after manufacturing, demonstration of coverage, repeatability etc. (V).
8. Assessment of mechanisms and mode of reactor vessel failure due to PTS including high and low pressure (depressurised) conditions (H).
9. Assessment of reactor pressure vessel mechanical response and the mechanical integrity of reactor (V).

V DETAILED INFORMATION ON THE SELECTED RPV DESIGN FEATURES AND PTS MANAGEMENT – THERMAL-HYDRAULICS ASPECTS

1. Thermal-hydraulic data:
    - Primary and secondary sides mass flow rates distribution among the loops and the reactor core, pressure drops and core delta T
    - Mass flow rates of all normal operation and safety injections against pressure (HA, HHSI, HPSI, Makeup system, LPIS, Charging system).
    - Emergency water tanks: volume & water temperature, set points
2. Design parameters of the RPV geometry data/dimensions including drawings (V)
3. Capabilities of the RCS depressurization and emergency core cooling systems including: diameter, length and layout of emergen-cy gas removal piping from the RPV, from SG headers, and from the pressuriser to the bubble tank (or the nominal flow rates in these pipes under nominal pressure difference conditions); possibility to open the pressuriser safety valves or power operated relief valve at pressures lower than their nominal opening setpoints (H&V).
4. Fuel element and fuel assembly data, fuel characteristics (structure description, dimensions, flow areas, thermo hydraulic data (mass flow rates, pressure and temperature drop), thermal-mechanical data of the fuel element composing materials against the temperature (thermal capacity and thermal conductivity) (V).
5. Core and kinetics data for both beginning and end of the fuel cycle (reactivity coefficients, peaking factors, initial core reactivity margin, etc.) (V).
6. Setpoint for starting the secondary side depressurization in case of blackout (is it outlet temperature from the core e.g. 650°C or water level in the SG e.g. 1.2 m) (H&V).
7. List of all protective and ESFAS actuation set points (V).
8. Failure temperature limit (v).
9. Technical possibilities for heat removal from the RPV wall (is there an agreed model for heat removal and with what parameters?) (H).
10. Information on the type of PORVs (Is the case of failure to close considered? If yes, what effective opening area of the PORVs is assumed?) (H).

11. Documents on material properties (chemical composition of the base material and the welds, mechanical properties including specified critical values, embrittlement characteristics). The material properties request applies also to the surveillance program samples (H&V).
12. Parameters relating to emergency coolant water availability change over time during an accident (H).
13. A description of the available results (with an explanation of the initiating event, all relevant system/hardware failures, all relevant human actions/inactions for each such severe accident, and the final containment status) (V).
14. Identification of accident scenarios and circumstances, which preclude primary depressurisation and a description of how high pressure conditions are addressed by the PTS management (apart from attempts to depressurise) (H).
15. A description of the analyses and available results (with an explanation of the initiating event, all relevant RPV Integrity and PTSA related system/hardware failures, which have been considered, all relevant human actions/inactions for such events, and the final RPV status) (V).
16. Analytical and experimental basis for modelling the timing and modes of PTS sequences related to high pressure RPV failure (including a discussion of the physical processes which lead to a delay between melt ejection and hydrogen combustion) (V).
17. Documentation concerning the verification and validation of the nodalisation of the coolant system and RPV models used for Temelín accident progression calculations and relating to thermal-hydraulic, thermal and load structure analyses and fracture mechanics simulation. (V)
18. Specifics for estimating the timing and rate of heat transfer including consideration of uncertainties (codes, nodalisation, assumptions, modelling choices, etc.) (V).
19. Model(s) (including changes to the model) and assumptions employed for estimating dynamics of heat transfer for shock cool-down of the RPV, including consideration of uncertainties (V).
20. Model(s) (including changes to the model) and assumptions employed for estimating dynamic loads resulting from thermal-hydraulic loads due to emergency operation and accident sequences progression, including the effects of impulse loads and thermal effects on the RPV wall (V).
21. Dynamic load factors used to describe the effects resulting from transient pressure loads (as contrasted with static loads) due to unexpected plant behaviour (V).
22. Identification of codes and parameter values used in modelling to simulate PTS events (V&H).
23. Spatial locations of sensors in the containment needed for PTS and RPVI (including the elevation and spatial location of the LBB and other sensors within the containment) (H&V).
24. Discussion of the availability of instrumentation in the containment to detect and localize leakage from the primary coolant loop and the RPV in particular. (H).

VI. OPERATION DOCUMENTATION ABOUT PTS/RPVI MANAGEMENT: ADMINISTRATIVE ARRANGEMENTS AND PROCEDURES (VLI I)

1. The overall emergency response organization and division of responsibilities in case of severe accidents in Temelín NPP (H&V).
2. The list of all administrative procedures relating to the planning, performance, and evaluation of the PTS related SAMGs (H&V).
3. Titles and overview of plant procedures that specify the responsibilities and authorities for key decisions (transition to EOPs, determination of strategies to be implemented under accident conditions, procedures used, termination of the strategy after it has been implemented, special approvals for implementation of strategies involving intentional fission product releases, long term recovery and termination of EOPs after a controlled state is achieved). Preferable way of providing information would be a matrix showing various key activities and/or accident phases and responsibilities for monitoring, evaluation, decision making, and implementation/actions (H&V).
4. Title and overview of the procedure(s) the roles of staff members in the implementation of strategies under conditions (H).
5. Description of how to accommodate briefing of incoming safety engineers on RPV and overall plant status (V).
6. Means by which to prioritize the restoration (either due to the initiating event or subsequent reasons) (V).
7. Identification of the preferred sequence of strategies in a PTS event, including the basis (e.g., sequence, etc.) (V).
8. Description of how the PTSA addressed the effects of a severe accident? (V).
9. Explanation of how to address differences between units (V).

VII OPERATION DOCUMENTATION ABOUT PTS/RPVI MANAGEMENT: TECHNICAL GUIDELINES, PROCEDURES OR INSTRUCTIONS (VLI I + II)

1. The PTS related list of symptom oriented EOPs implemented at Temelín NPP (H).
2. Structure of the PTS Guidelines package for Temelín NPP (list of EOP guidelines/procedures, diagnostic logic trees, computational aids, etc.) (H).
3. Procedures for establishing embrittlement conditions under which RPV have been qualified (H).
4. Procedures for maintaining relevant equipment during PTS avoidance and mitigation actions or functions for the equipment's anticipated life (H).
5. Discussion of training aids used in training control room personnel, managers with accident management responsibilities, and technical support centre personnel concerning PTS accident progression (V).
6. Assessment of means to follow up on PTS avoidance and mitigation actions or functions during core degradation (H).
7. An indication of whether the EOPs anticipate that operator actions would be taken directed towards PTS avoidance and mitigation actions or functions. (V).
8. Identification of the symptomatic cues to the control room operators to begin observing PTS/RPVI instrumentation (V).
9. Current EOPs provisions related to primary depressurization, steam generator water level management, core-cooling management, together with their background documentation, if not yet implemented, then the latest development version reflecting their current developmental status (V).

10. Setpoints and criteria used to initiate, throttle, terminate, or preclude initiation of PTS avoidance and mitigation actions or functions (including functions anticipated in the EOPs), including method of incorporating uncertainty into the setpoints and criteria (V).
11. Description of the requirements for and means to accomplish overrides, defeat interlocks, or block automatic protection signals in case required by the PTS avoidance and mitigation strategies (V).

VIII OPERATION DOCUMENTATION ABOUT PTS/RPVI MANAGEMENT: PTS RELATED EMBRITTLEMENT RECORDS, REPORTS AND CURRENT STATUS OF SAMPLING (VLI I + II)

1. List of reports related to neutron flux and spectra prediction for surveillance specimen and reactor pressure vessel wall (H&V).
2. List of reports related to irradiation (neutron fluence) prediction, verification and evaluation (H&V).
3. List of reports for material qualification considered (H).
4. Sample(s) of records for RPVI, tests and/or analysis with supporting documentation (H).
5. Current status of PTS and RPVI measures development/implementation at Temelín NPP and schedule for future activities (H).
6. List of training topics/courses performed on PTS, number of staff covered (H).
7. Sample(s) of training material and records from training(s) performed on PTS avoidance and mitigation (H).
8. Experimental results of surveillance sample testing information on the currently irradiated set of surveillance samples (H).
9. Status of the NDT qualification program, volume and results of qualified RPV testing (H).

IX EVALUATIONS BY THIRD PARTIES

1. Evaluation/review reports or report sections prepared by any independent reviewers (authorities, organizations or experts of other countries) on the issues related to PTSA and RPVI for Temelín NPP, if available (H&V).
2. IAEA Reviews of PTS and RPVI as well as related topics also with respect to safety issues status (H&V).

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[1] [IAEA Report WWER-SC-171 “Review of WWER-1000 Safety Issues: Resolution at the Temelín NPP, IAEA, Vienna, 1996

[2] US NRC “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants”, NUREG-0800, Washington DC, 1981

[3] IAEA Safety Standards Series No. NS-R-1 “Safety of Nuclear Power Plants: Design”, IAEA, Vienna, September 2000.

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[6] IAEA Guidelines on Pressurised Thermal Shock Analysis for WWER Nuclear Power Plants, IAEA-EBP-WWER, IAEA, Vienna, 1997

LIST OF GUIDANCE DOCUMENTS (applicable and/or applied):

[1] Radiation Embrittlement of Reactor Vessel Materials, US Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, 1998

[2] ASTM Standard E 185-82, Conducting Surveillance Tests for Light-Water Cooled Power Reactor Vessels, Annual Book of ASTM Standards, Vol.12.02, American Society for Testing and Materials, 1993

[3] KTA Sicherheitstechnische Regel, Überwachung der Strahlenversprödung von Werkstoffen des Reaktordruckbehälters von Leichtwasserreaktoren, KTA 3203, Fassung 03/84

[4] PNAE G-7 002-86, Standards for Strength Evaluation of Component and Piping of Nuclear Power Plants, Energoatomizdat, Moscow, 1989

[5] SÚJB Instruction, Instruction and Recommendations for Lifetime Assessment of VVER RPV and Reactor Internals during NPP Operation, SÚJB 1998

[6] SÚJB Standard, – Residual Lifetime Assessment of VVER Nuclear Power Plants Components and Piping, Standard Technical Documentation of Association of Mechanical Engineers, Section IV. – SÚJB

[7] ESIS Procedure for Determining the Fracture Behaviour of Materials, ESIS P2-92, 1992, University of Delft

[8] Reactor Vessel Material Surveillance Program Requirements, Appendix H to Part 50 of Title 10 of the Code of Federal Regulations

[9] Conducting Surveillance Test for Light-Water Cooled Nuclear Power Reactor Vessels, Annual Book of ASTM Standards, E 185

[10] Specification for Steel Forgings, Alloy, for Pressure and High Temperature Parts, Annual Book of ASTM Standards, A 336

LIST OF AVAILABLE BACKGROUND DOCUMENTS (considered applicable and/or applied):

[1] Ballesteros, H. Ait Aderrahim, L. Debarberis, T. Lewis, C. Sciolla, M. Valo, Neutron Calculations in European Reactors, AGE-MADAM(98)-D003(1998)

[2] A. Deardorff, Example problems to assess flaw acceptance margins, Committee correspondence, Presented at ASME IX, Working group on flaw evaluation, Meeting, Kansas City, May 6,1998

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[14] L. M. Davis, A Comparison of Western and Eastern Nuclear Reactor Pressure Vessel Steels, AMES Report No. 10, Catalogue Number: CD-NA-17327 EN-C, Brussels-Luxembourg, 1997

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ANNEX F

PRESENTATION OF MONITORING RESULTS BY THE AUSTRIAN EXPERTS' TEAM ON THE OCCASION OF THE CZECH-AUSTRIAN BILATERAL MEETING

on November 30, 2004,
in Dolni Dunajovice, Czech Rebublic

IRR / ARCS

Temelín Roadmap Process Specialists' Monitoring

Roadmap Project PN9:

Item 3 Reactor Pressure Vessel Integrity and Pressurised Thermal Shock Results Evaluation and Findings

Presentation prepared for the Bilateral Meeting in Dolni Dunajovice, November 29/30, 2004

G.H. Weimann, I.Tweer

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Melk Process Result

With regard to this item Reactor Pressure Vessel Integrity and Pressurised Thermal Shock Issues the Roadmap for the implementation of Annex I and II reads as follows:

"Objective: The reactor pressure vessel (RPV) integrity under pressurised thermal shock (PTS) conditions shall be maintained with a ***sufficient safety margin against brittle fracture throughout the NPPs service life****.*

Present status and Specific Actions Planned: The NPP Temelin is commissioned respecting pressure-thermal (PT) curves calculations developed according to Westinghouse methodology. These calculations will be expanded with set of further PTS analysis for both units using a step by step approach ***with full respect of the IAEA Guidelines*** *for the PTS analysis. The PTS analysis will be finished in accordance with approved project plan for the item".*

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Reactor Pressure Vessel Integrity (RPVI) and Pressurised Thermal Shock (PTS)

Workshop Coverage

1. **PTS analysis**
2. **Surveillance program: RPV (reactor pressure vessel) material embrittlement monitoring**
3. NDT (non-destructive testing) program ➡ PN 10
4. ISI - In-Service inspections ➡ PN 10
5. **Mitigative core design / operation provisions**

Development of PTSA Codes and Requirements

- **Russian design: Russian Code regulations PNAE-G-7-002-86 [PNAE 86]**
- **Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants, IAEA-EBP-WWER-08 [IAEA 1997]**
- **Construction and licensing by Czech National Authorities: Instructions and Recommendations for Lifetime Assessment of VVER RPV and Reactor Internals during NPP Operation, ASI Code, Section III and IV – being developed [SÚJB 1998]**
- **Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs, VERLIFE, accepted by SÚJB 2004 [VERLIFE 2004]**

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Austrian Experts' Team
Main Conclusions (1)

1. General Procedure

Significant reductions of conservatisms:

- **Reduced postulated crack depth under the assumption of qualified NDT:**
  [IAEA 1997] up to smaller s/4
  [SUJB 1998] up to s/8
  [VERLIFE 2004] up to s/10
- **Elimination of the safety factors for SIF calculations in case of crack depths <s/4:**
  [IAEA 1997]: n_k=2 or √2 ΔT= 10[K]
  [SUJB 1998]: n_k=1 ΔT= 0[K]
  [VERLIFE 2004]: n_k=1 ΔT= 0[K]
- **Application of the WPS (warm pre-stressing) effect:**
  not allowed: [PNAEG 86] and [SUJB 1998]
  allowed: [IAEA 1997] 80 % criterion
  [VERLIFE 2004] 90 % criterion

The Austrian Experts would appreciate to know the scientific reasoning for these changes.

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Item 3 RPV and PTS Results Evaluation and Findings
Bilateral Meeting in Dolni Dunajovice November 2004 5

Austrian Experts' Team
Main Conclusions (2)

2. PTSA procedure

- **All important accident groups have been evaluated, ETE-PTSA is of a completeness not achieved so far for WWER-1000.**
- **The used computer codes for TH/mixing calculations are considered to meet the actual state-of-the-art for simulation and validation.**
- **The most critical accident conditions defined for the transients must be checked for compliance with [IAEA 1997]. For the group PRZ SV the total loss of offsite-power has not been included in the presented results.**
- **Some accidents (PSV43) simulations have not been extended to the point of WPS applicability.**
- **Several accident transients (PSV41) need „guaranteed" operational procedures to avoid RPV failure as a consequence of PTS loads.**

Remaining question:

Which criteria have been developed to establish an answer to the safety relevant question of reduction in conservatism?

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Item 3 RPV and PTS Results Evaluation and Findings
Bilateral Meeting in Dolni Dunajovice November 2004 6

Austrian Experts' Team
Main Conclusions (2 cont'd)

(2) PTSA procedure

- Heat transfer calculations appear to be inconsistent. Assumptions regarding heat transfer mechanisms and heat transport properties remained unclear.
- The computer codes applied and FE models for temperature and stress field calculations are considered state-of-the-art.
- The most recent approach of including the cladding into the FE model involves an additional reduction of conservatism.
- The NDT inspections carried out up to now cover only part of the ISIs required, in particular concerning the ECT. NDT must prove the absence of cracks relevant for PTS – cladding cracks and underclad cracks – as required for RPVI demonstration.

Remaining questions:

- How is conservatism of the heat transfer assumptions demonstrated?
- How can the mandatory assumption of an intact cladding be demonstrated?
- How will the completeness required for RPVI demonstration be achieved?

Austrian Experts' Team
Main Conclusions (3)

3. PTSA result

- [IAEA 1997] safety factors recommended for postulated crack sizes < s/4 and the 80% WPS criterion applied to a transient presented at the Prague Workshop:
  the resulting value T_k^a = 66,3 [°C] would be reduced to T_k^a = 15 [°C].
- Applying the [SUJB 1998] limit criterion

  $$T_k^a = \min\{T_k^a{}_j\} - \delta T, \text{ with } \delta T = 14 \text{ [K]}$$

  to the PTSA results from the Rez Workshop

  $$\min\{T_k^a{}_j\} = 64{,}6 \text{ [°C]}$$

  (calculated according VERLIFE, i.e. without the safety factors as defined by IAEA and applying the 90% WPS criterion).

  yields T_k^a = 50,6 [°C]

Remaining question:

Is there a "sufficient safety margin against brittle fracture throughout service life"?

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Main Conclusions (4)

4. Surveillance program

- The modifications implemented at ETE for the surveillance program make it the first one to allow obtaining reliable surveillance data for WWER-1000 RPV-steels.
- The embrittlement coefficient A_F values specified in the Russian Code cannot be considered conservative.
  Evaluation of surveillance and research results published shows this.
- Differences of initial T_{k0} values for different electrode heats have been observed (weld No.4 in ETE-1).
  Surveillance samples of the more critical material are irradiated in ETE-2 only.
  The same is true for HAZ samples with even higher T_{k0} values.

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Item 3 RPV and PTS Results Evaluation and Findings

Bilateral Meeting in Dolni Dunajovice November 2004

9

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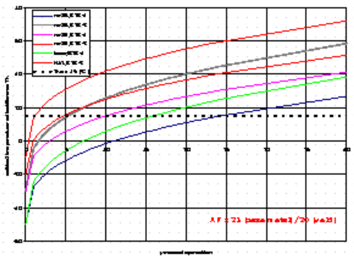
Main Conclusions (5)

5. Predicted embrittlement curves Austrian Experts' Assessment:



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Item 3 RPV and PTS Results Evaluation and Findings

Bilateral Meeting in Dolni Dunajovice November 2004

10

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Main Conclusions (6)

6. Embrittlement management:
Reduction of Neutron Radiation to the RPV-Wall

Remaining questions:

- **Was the OUT-IN strategy introduced for irradiation embrittlement mitigation or is it just a side effect of power output optimization?**
- **Which modifications of the Westinghouse core design, - the first of its kind to be qualified, - are suitable for fluence minimization at the RPV wall?**

Reactor Pressure Vessel Integrity (RPVI) and Pressurised Thermal Shock (PTS)

Issues not touched

RPVI relevant components:
- **Main coolant line penetrations**
- **Vessel head and other penetrations**
- **Main flange tightness**
- **Internals**

RPVI relevant environment:
- **Coolant chemistry**
- **Hydrogen diffusion**
- **Corrosion**
- **Fatigue**

RPVI relevant integrity issues:
- **Surveillance measures ascertaining LBB applicability**
- **RPVI verification**
- **RPVI severe accident behaviour**

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Reactor Pressure Vessel Integrity (RPVI) and Pressurised Thermal Shock (PTS)

Concluding Remark

- Monitoring has concentrated on the consistent application of RPVI demonstration from the early design states to the establishment of the license for operation including the provisions and prerequisites for continuing RPVI verification and RPV material surveillance.
- The monitoring result indicates that
  - Worst case conditions for thermal transients might have to be revisited,
  - Remarkable safety factor reductions were introduced for PTSA,
  - The RPV material surveillance program is in an advanced state of implementation, reliable results can be expected. However, to obtain a statistical endorsement a sufficiently ample database of the materials' embrittlement is not yet available.

Therefore the Austrian Experts' Team recommends to the Government of Austria to continue further monitoring of the related issues.

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Item 3 RPVI and PTS Results Evaluation and Findings

Bilateral Meeting in Dolni Dunajovice November 2004 13

Thank you for the attention!

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Reactor Pressure Vessel Integrity (RPVI) and Pressurised Thermal Shock (PTS)

Issues to be considered with Safety Factors

1. The number of accidents with RPV thermal shock potential is considerable
2. Conservatism of fracture mechanics assessment not proven
3. Thermal hydraulics uncertainties don't allow perfect transient modelling
4. Mixing calculation solvers become very unstable during simulation
5. Remarkable uncertainties with material properties
6. Broad uncertainty of fluence determination
7. Reliance on correct and in time actions by the operator in following EOP's
8. NDT-methods applied in ISI are likely to miss deficiencies
9. Entire cladding must be completely free from unacceptable deficiencies
10. Certain areas of the RPV-wall cannot be inspected.

ANNEX G

ADDITIONAL INFORMATION:
THERMAL HEAT TRANSFER CONSERVATISM IN PTSA

(SIMULATION CALCULATIONS FOR LICENSING IN THE CZECH REPUBLIC)

Thermal Conservatism credibility explanation document

Pavel Kral, Petr Muhlbauer, Vladislav Pistora, NRI Řež – Vienna, October 2005

1) Additional comments to conservatism of TH analyses (with regard to your questions about conservatism of HTCs): *A very extensive set of conservative presumptions has been applied in the system TH analyses for PTS/ETE (about 40 conservative assumptions). Especially the very detailed and conservative modelling of ECCS systems (including plugs of cold water, heat exchanger with minimal temperature of technical cooling water (5 °C) etc.) results in extremely low temperatures of water injected from ECCS into primary system.

   * Remix presumes total stagnation of basic flow in loop as soon as the criterion for stratification is true, which results in artificially faster DC cooldown (both plume and ambient).

   * Remix does not model side moves of the cold plume (that can be seen in some CFD results and leads to less adverse results).

   * When comparing the values of HTC from 2-D calculation of DC by Relap5 to the approximate constant values in IAEA methodology (Appendix III) for PTS (5000 W/m^2K for plume, 2000 W/m^2K for ambient), one can see that the R5 results are in good agreement. In initial phases of process (SBLOCA, PRISE etc.), the plume HTC is little bit higher than 5000 W/m^2K (usually the maximum is at about 6000 W/m^2K for up to one thousand seconds), and in the later phase the HTC decreases (as temperature and velocity fields homogenise). When comparing the calculated courses of HTC in the ambient region, is usually lower than the "IAEA" constant 2000 W/m^2K, which is conservative (the lower values). The computed values of ambient HTC after RCP coast down (i.e. in natural circulation and stagnation phases) usually slowly decrease from 1500 …1700 W/m^2K to 1000 W/m^2K (see the point "3" and the attached file).

   * When comparing our TH analyses to the latest USA practice in PTS analyses (ISL, USNRC) where more or less only RELAP5 has been used by ISL for all the TH analyses (covering also the mixing part), one can see that we are more conservative. They used NRC version of RELAP5 with 2-D nodalization of reactor DC. It is the same approach as we have used in the system TH analyses for PTS, but we have recalculated all the cases by Remix consequently, which leads to more conservative results (similar asymmetry but faster cooldown).

2) Example of results of Remix test calculation against measured data from NPP Temelín See the attached file "Remix_Seidelberg.doc".

3) Example of HTC courses in electronic form See the attached file "HTC-SBLOCAetc.doc".

4) DC levels in some MB/LB LOCA See the attached file "DC-level-MB-LB-LOCA.doc". There are 4 examples of medium and large break LOCA with different break size, break location, reactor power and ECCS availability. For each case, there are 2 figures – a figure with system pressures and a figure with collapsed levels in reactor. The figures are not completely translated into English, but I hope, they will be "understandable". In the figures with level, the DC level is the violet dotted line (the "inner reactor" level is solid blue line) and there are also auxiliary lines depicting elevation of the inlet nozzles bottom and top. One can see, that the DC level depression into "core elevations range" is rare and if happens, then it is just short-term. Therefore, the problem of necessity of 3-D modelling of reactor DC in these cases (as you mentioned in discussions) is not so serious. Our 2-D modelling of DC behaviour with Cathare code is sufficient.

Teil 1

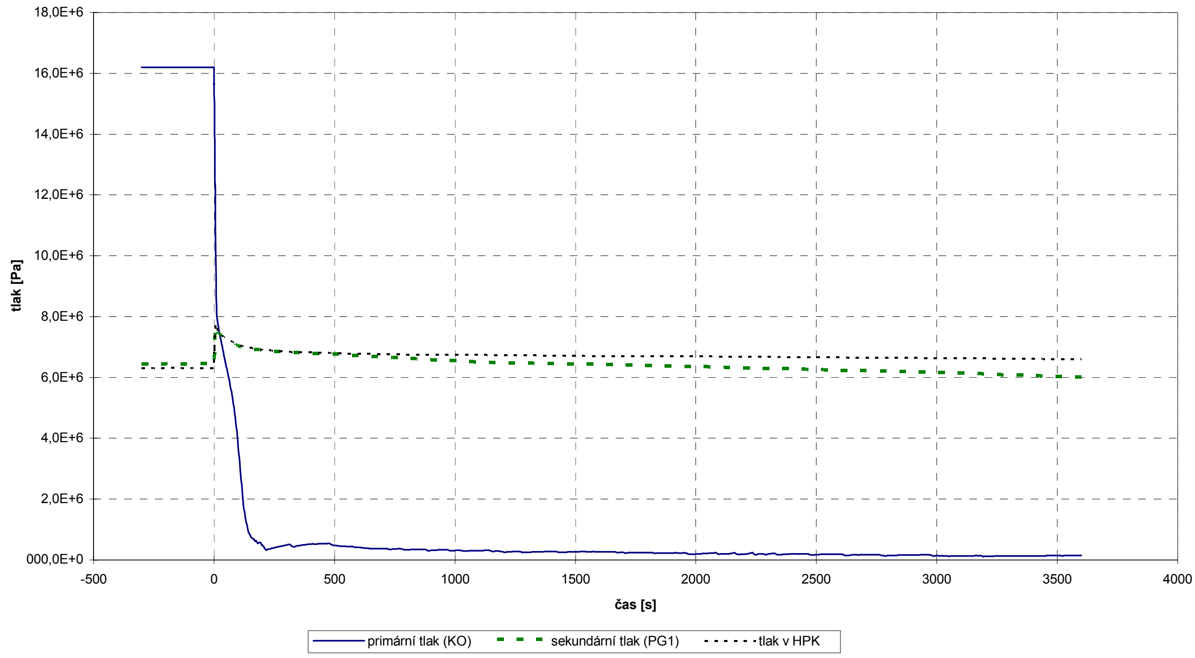

Fig. C300min-01: Primary and secondary pressure (break D300 in CL2, N100%, min. ECCS)

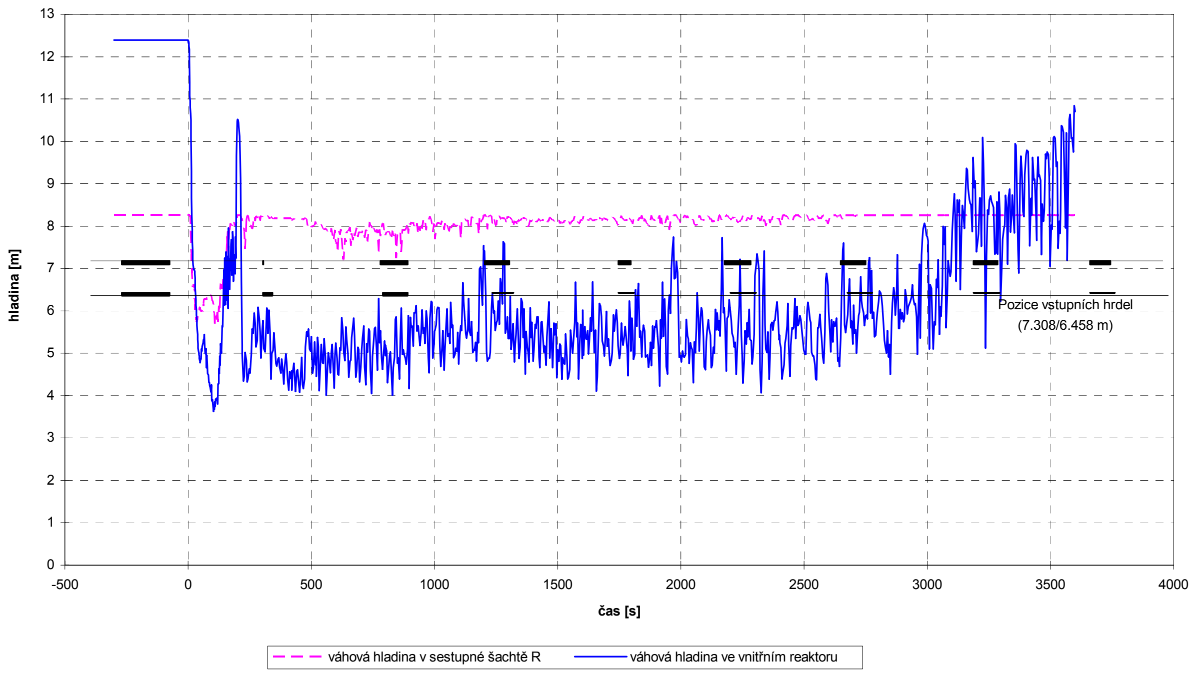

Fig. C300min-02: Collapsed levels in reactor (break D300 in CL2, N100%, min. ECCS)

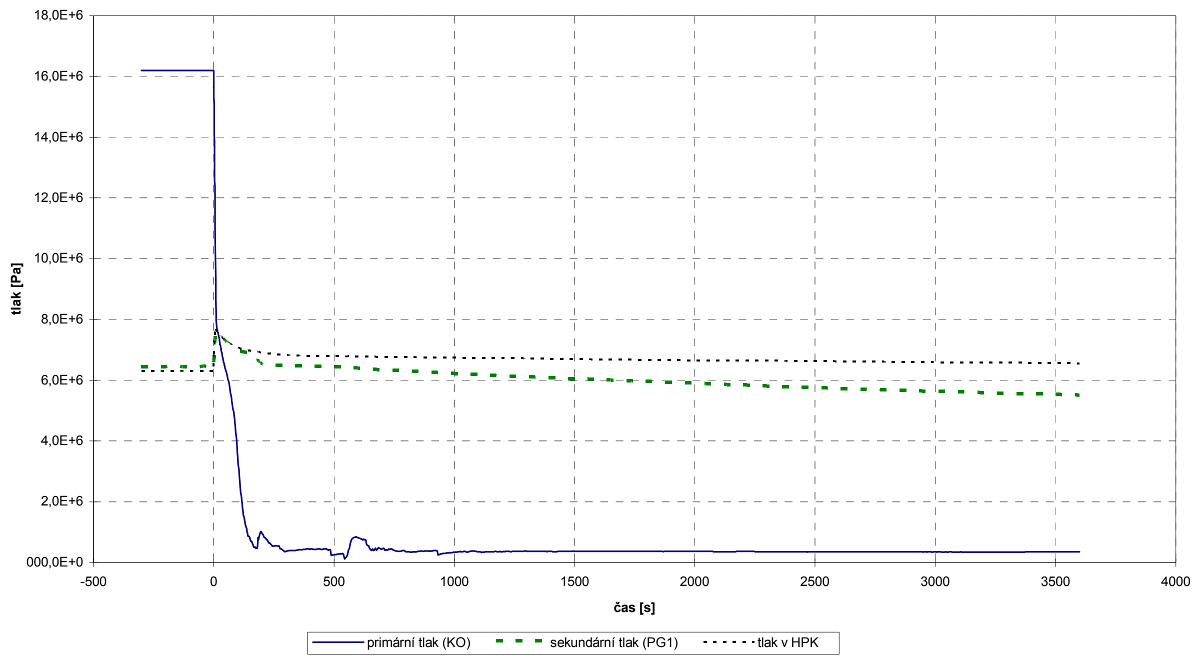

Fig. H300max-01: Primary and secondary pressure (break D300 in HL4, N100%, max. ECCS)

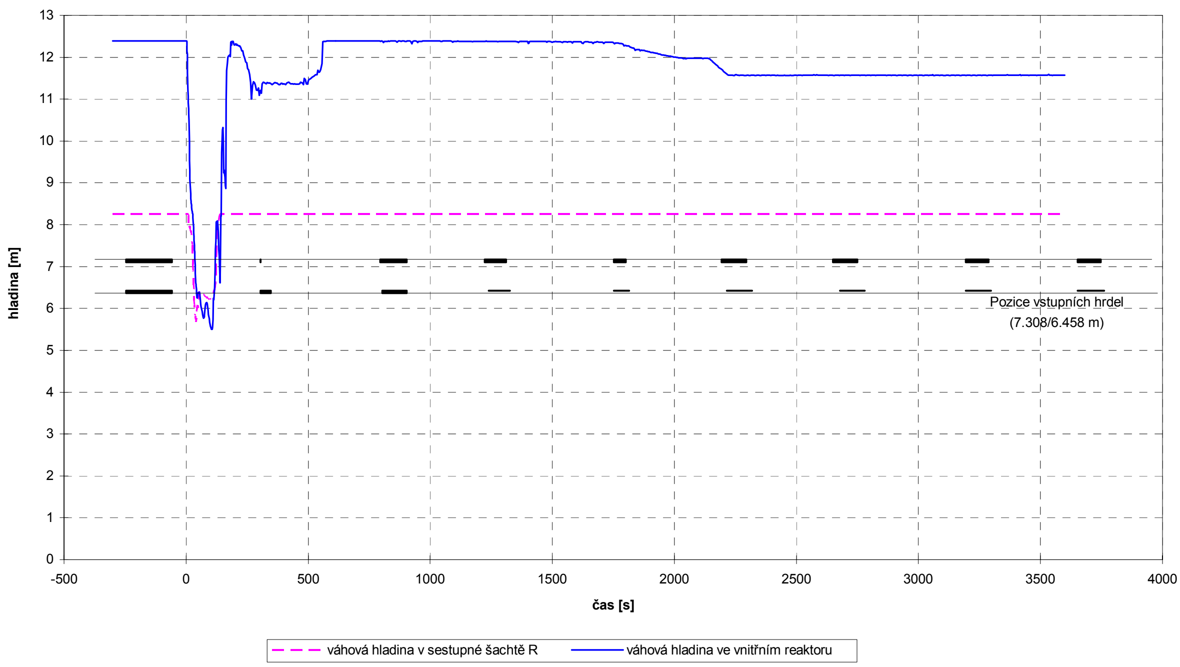

Fig. H300max-02: Collapsed levels in reactor (break D300 in HL4, N100%, max. ECCS)

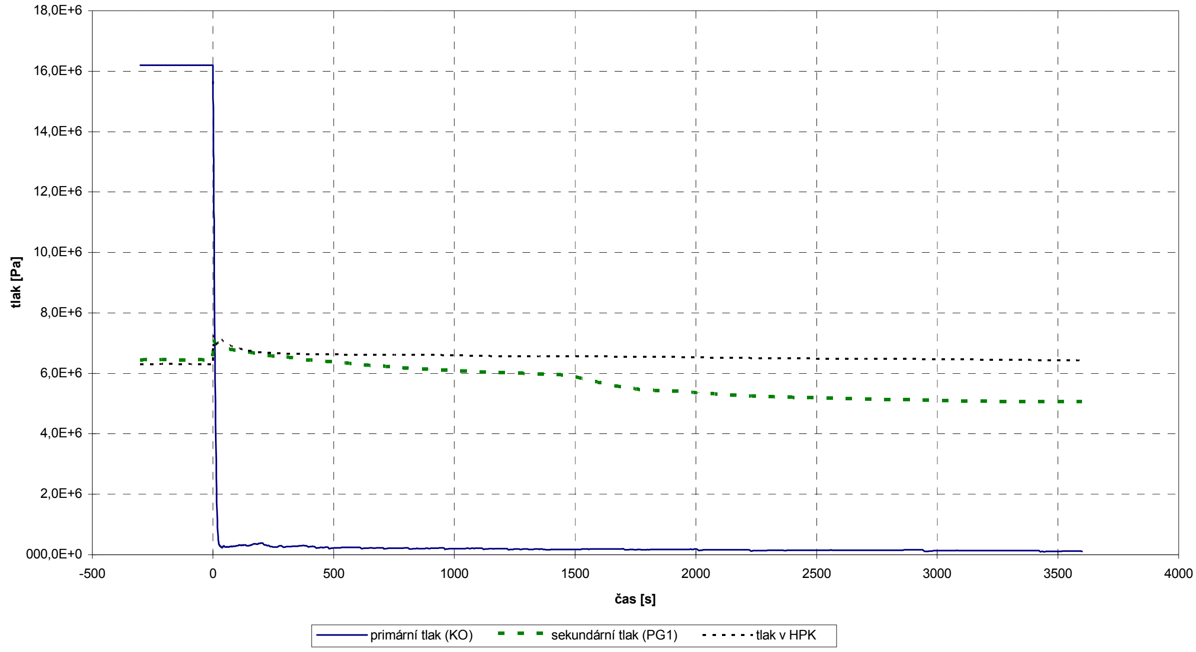

Fig. C850min-01: Primary and secondary pressure (break 2xD850 in CL2, N100%, min. ECCS)

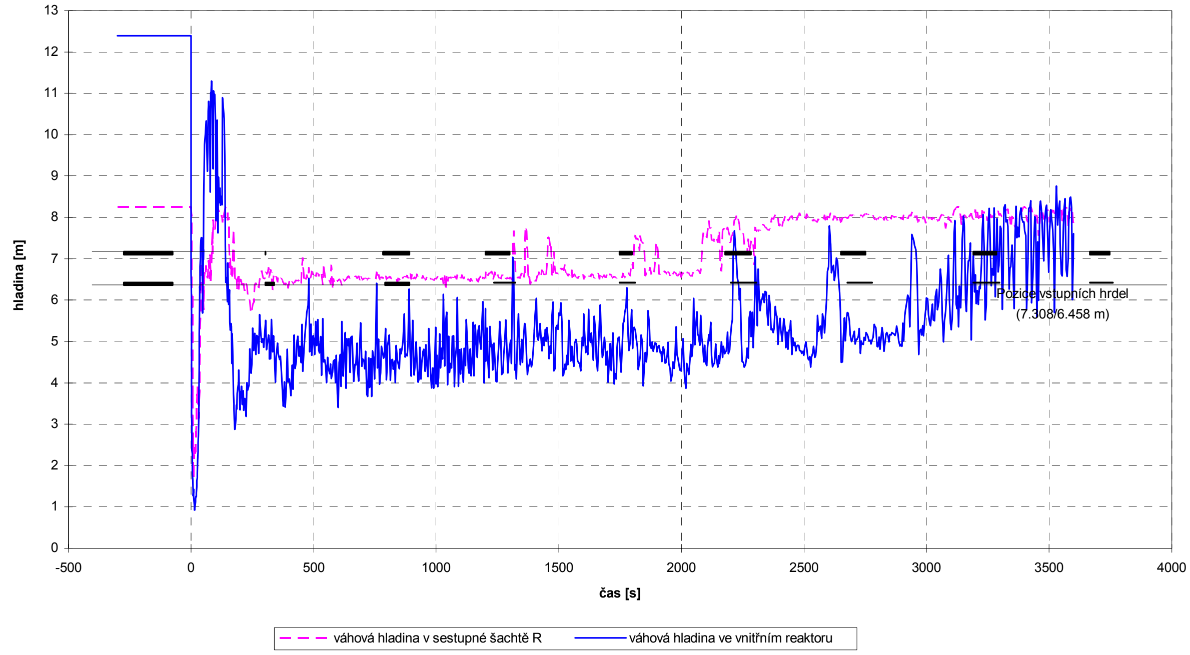

Fig. C850min –02: Collapsed levels in reactor (break 2xD850 in CL2, N100%, min. ECCS)

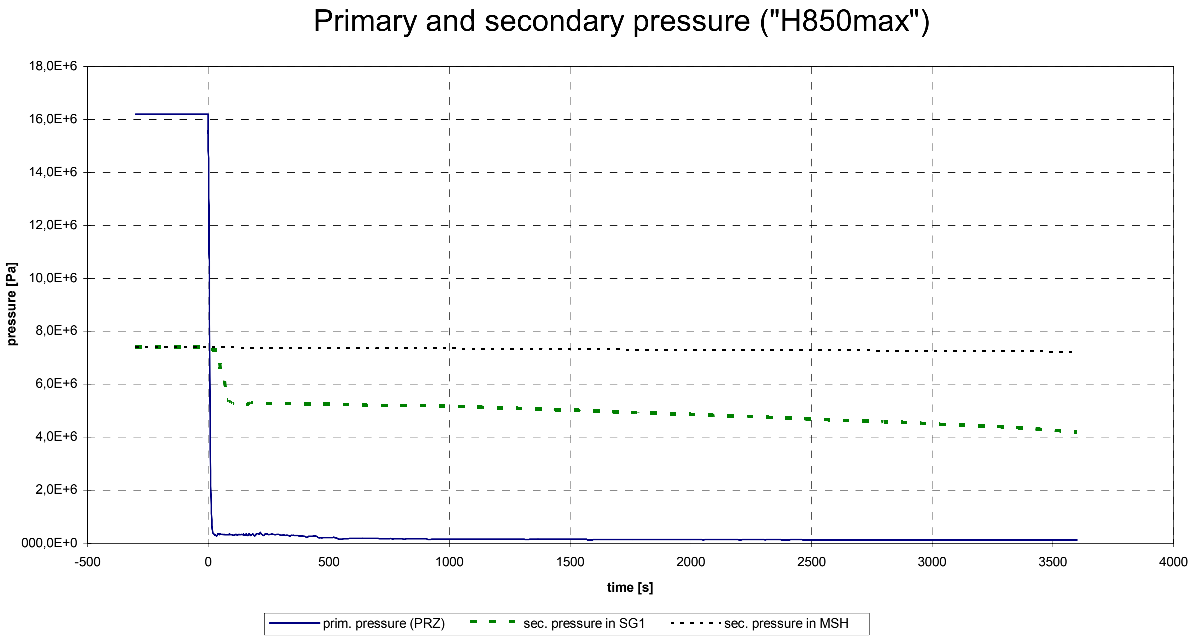

Fig. H850max-01: Primary and secondary pressure (break 2xD850 in HL4, N100%, max. ECCS)

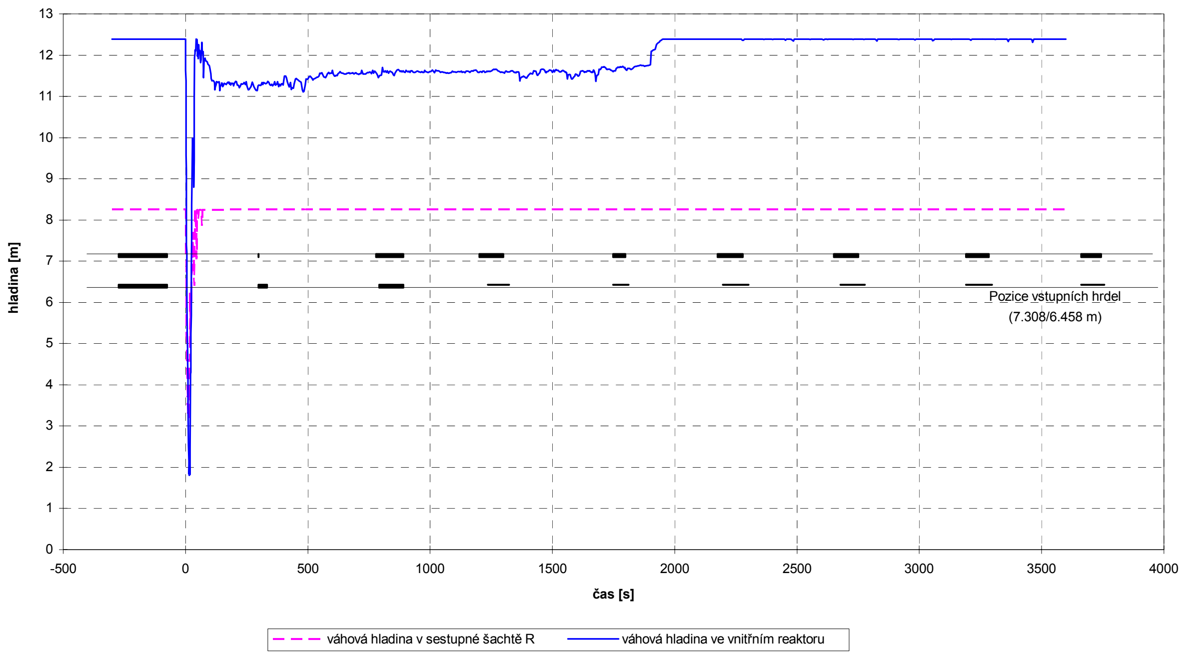

Fig. H850max –02: Collapsed levels in reactor (break 2xD850 in HL4, N100%, max. ECCS)

Teil 2

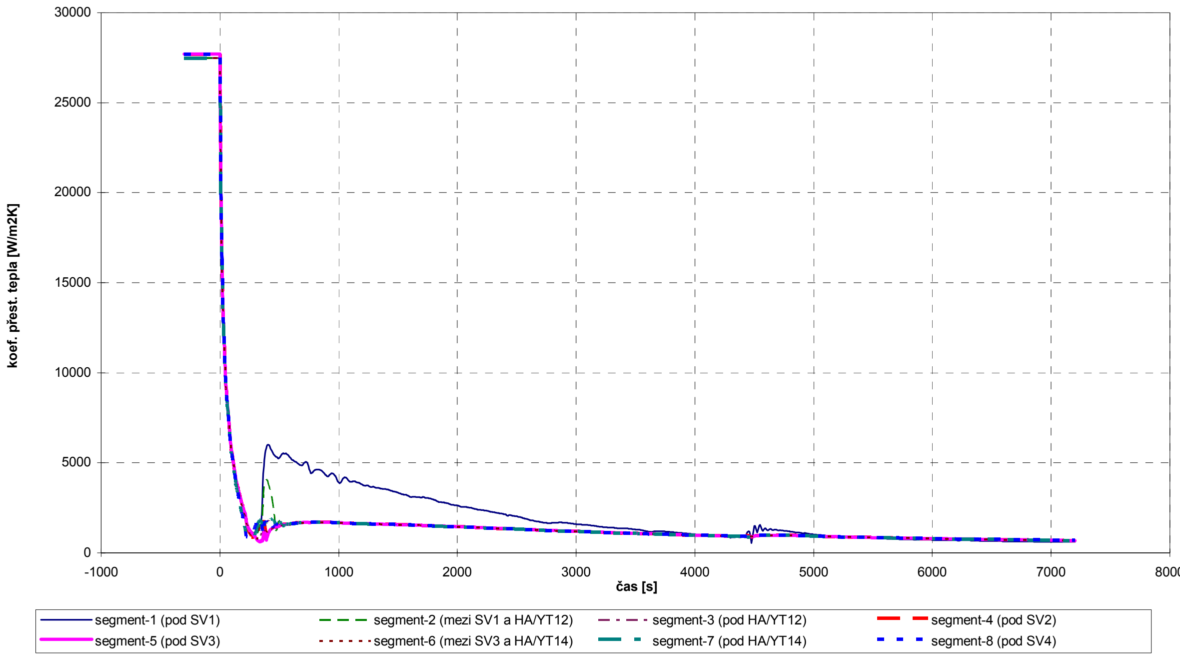

Fig. C32min-01: HTCs in layer-3 (at the elevation of top part of the core) (break D32 in CL2, HZP, min. ECCS)

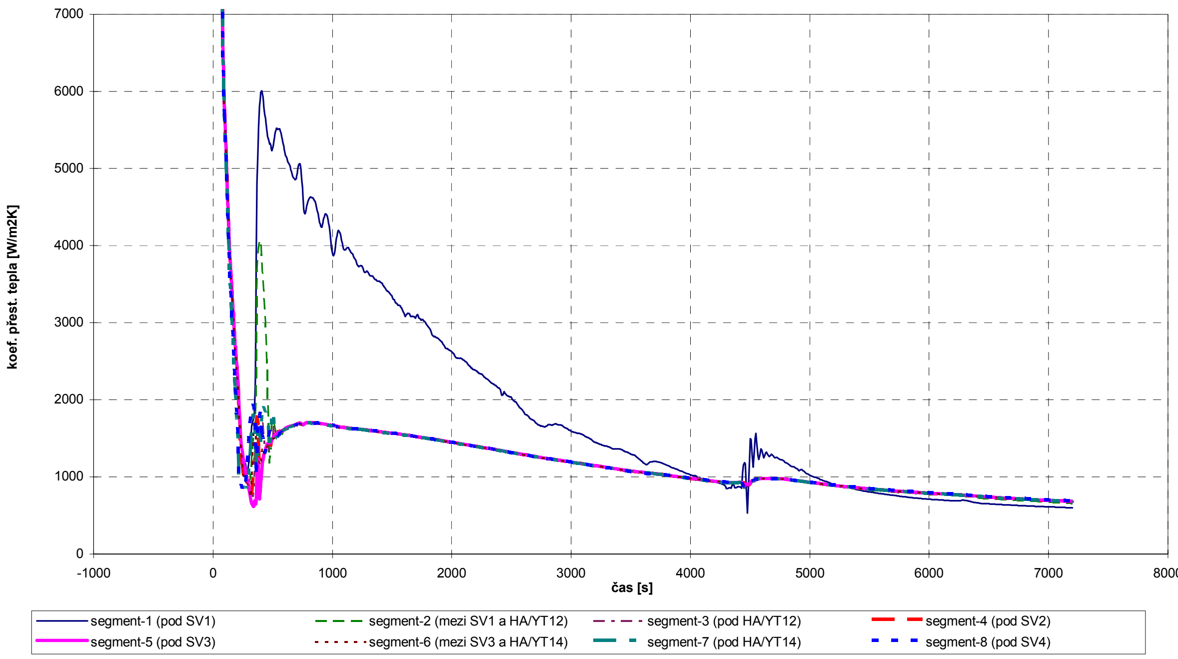

Fig. C32min-01: det HTCs in layer-3 (at the elevation of top part of the core) – DETAIL (break D32 in CL2, HZP, min. ECCS)

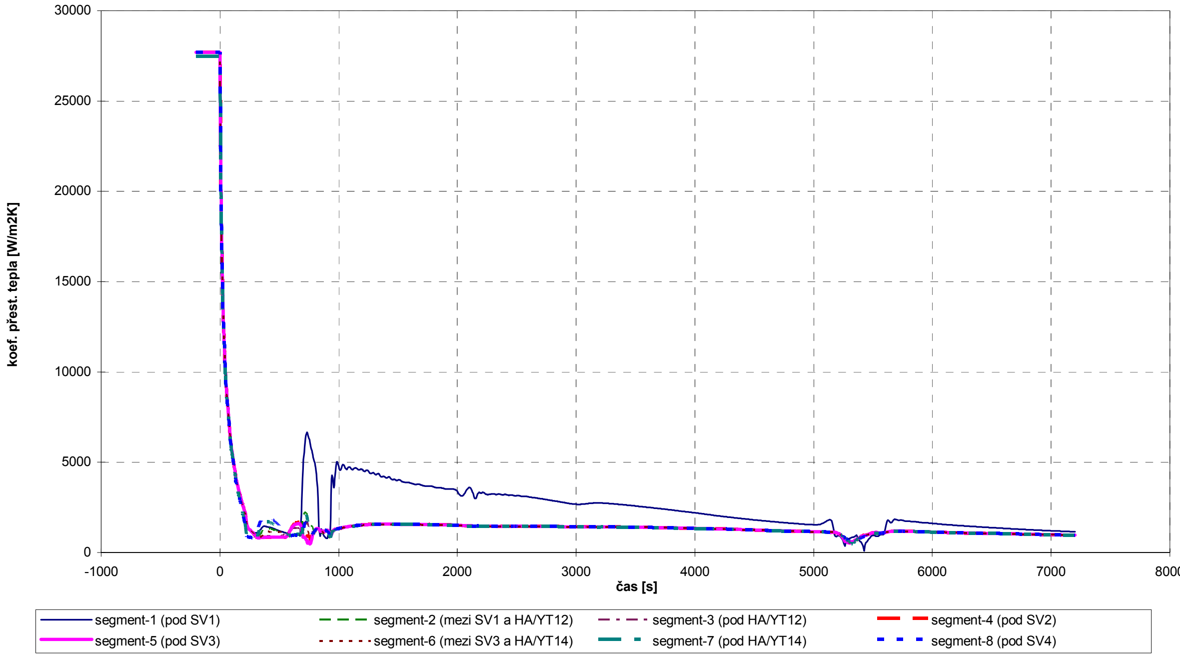

Fig. 3SGT-01: HTCs in layer-3 (at the elevation of top part of the core) (rupture of 3 tubes in SG1, HZP, min. ECCS)

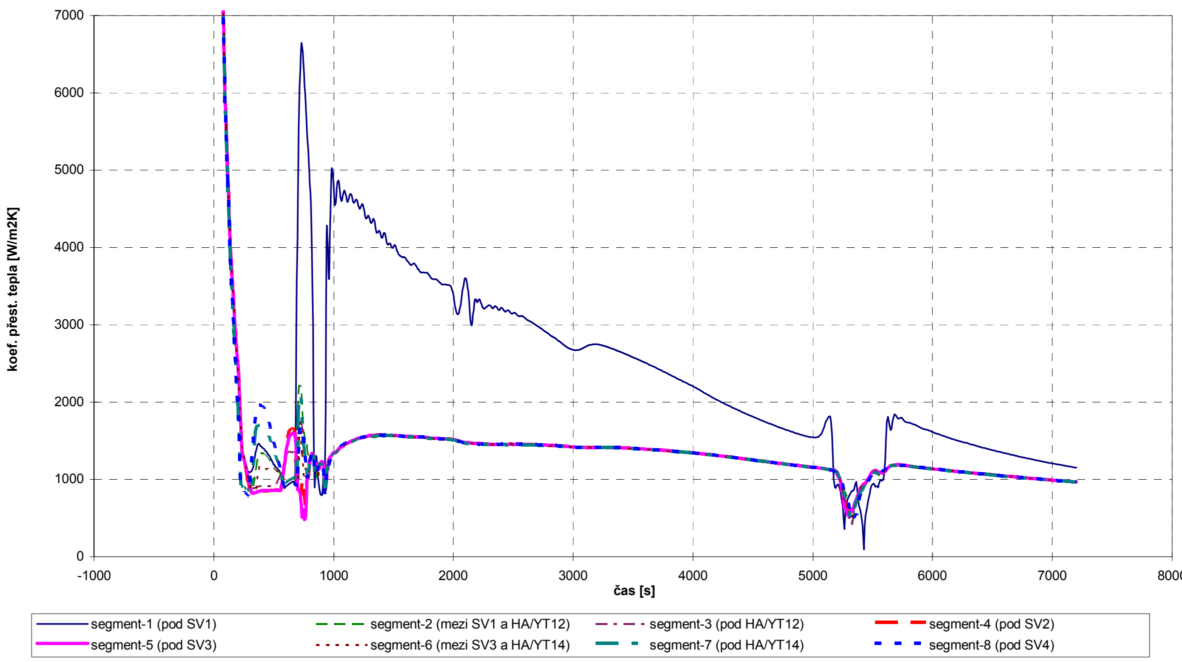

Fig. 3SGT-01: det HTCs in layer-3 (at the elevation of top part of the core) – DETAIL (rupture of 3 tubes in SG1, HZP, min. ECCS)

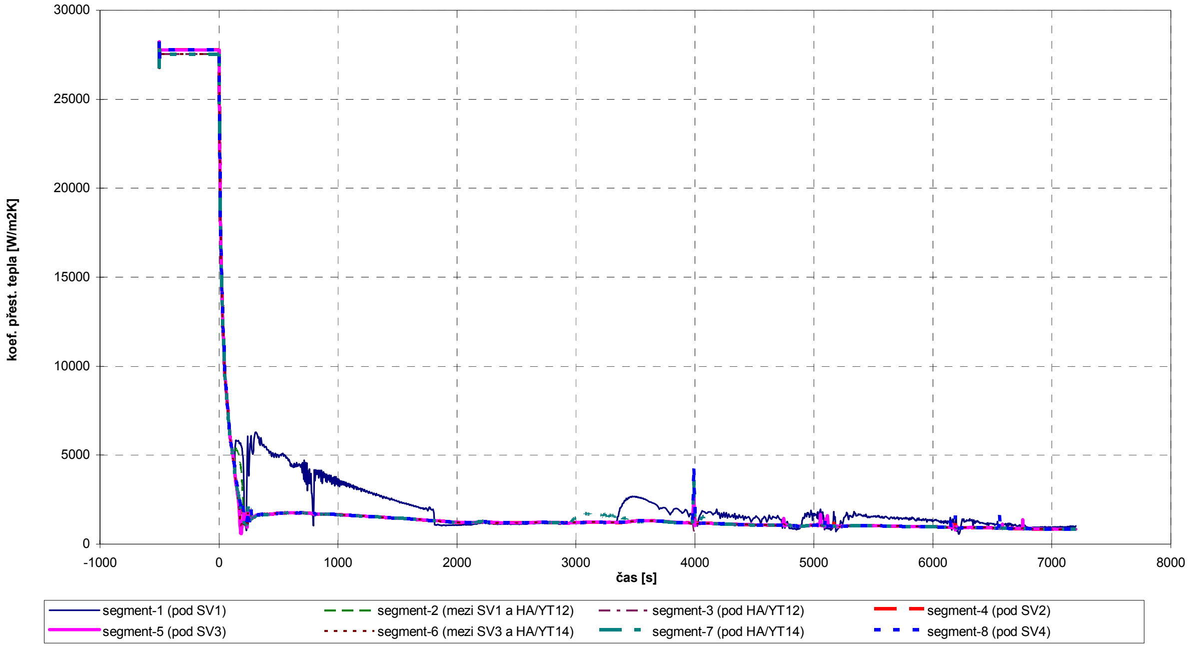

Fig. PSV41-01: HTCs in layer-3 (at the elevation of top part of the core) (inadvert. opening of PRZ SV, HZP, min. ECCS)

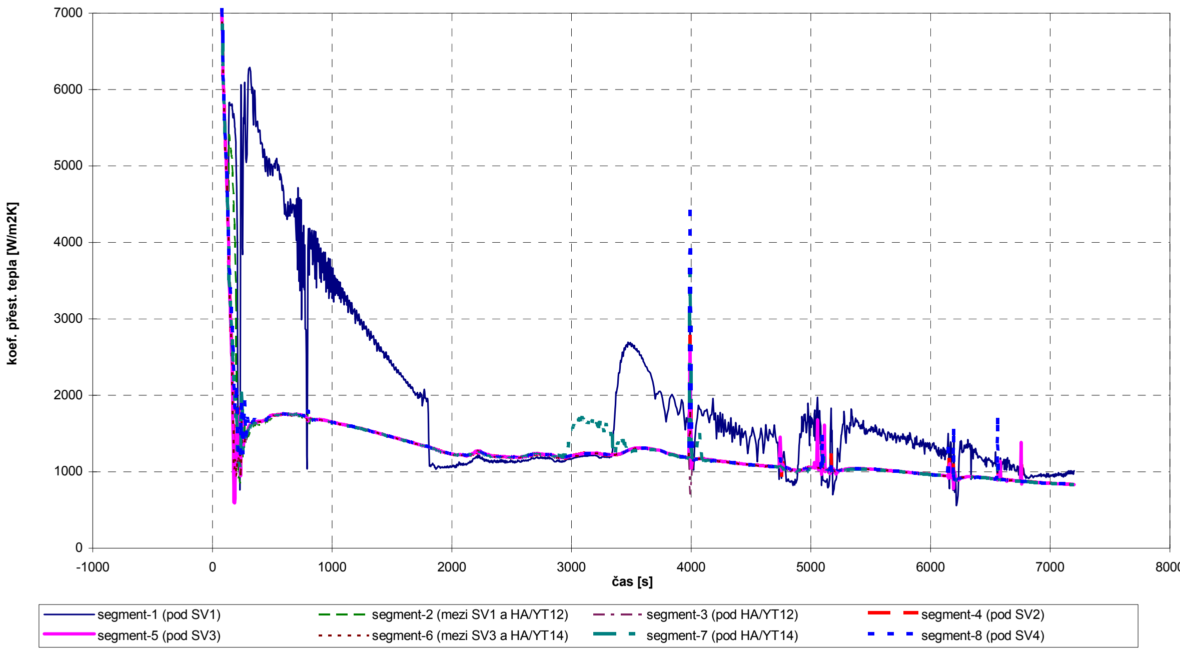

Fig. PSV41-01: det HTCs in layer-3 (at the elevation of top part of the core) – DETAIL (inadvert. opening of PRZ SV, HZP, min. ECCS)

ANNEX H

MISSION STATEMENT

AS ADOPTED BY THE EXPERTS' TEAM

Monitoring Mission Statement

The independent Experts' Team agreed on a "Mission Statement" to define the co-ordinated monitoring process.

"Monitoring" is a process performed in a predefined frame addressing selected Roadmap Items defined in the "Conclusions of the Melk Process" as well as in the "Roadmap" and the solutions to these Roadmap Items adopted by the Czech side.

Related issues and their solutions are monitored on the basis of reference safety criteria and requirements coherent with Safety Approaches accepted in Western Europe. The requirements are checked against the generally applied Defense in Depth Concept.

The Monitoring has the objective to obtain evidence that adequate solutions have been submitted by the licensee to the licensing authority and that these solutions have been appropriately evaluated and approved by the regulator. Monitoring aims at performing an evaluation of the quality and adequacy of an overall process and the implementation results.

The Czech side has offered documentation and discussion opportunities.

The Monitor, in order to form a consistent opinion should be provided with the opportunity to ask for additional information and evidence or request supporting assessments to understand the evidence presented.

Reports of the Experts' Team therefore include monitoring results of

What has been done,

How the applicable requirements have been addressed,

How the safety objectives' and requirements' compliance was analysed and justified for the proposed solutions, and

How were evaluated the solutions in the frame of the licensing process and considered in the related regulatory process

The Monitors were not tasked with performing a licensing review of Temelín NPP, and nothing in their reports may be construed to represent any such review. The responsibility for the safety and licensing of Temelín remains with ČEZ a.s. as the owner of the facility, and with the SÚJB, as the designated nuclear licensing and regulatory authority under Czech law, in correspondence with the related international requirements and practices.