umweltbundesamt

WORKSHOP WITH PLANT WALKDOWN IN TEMELIN, UNDER THE CZECH-AUSTRIAN BILATERAL AGREEMENT SEPTEMBER 26/27, 2006

Report − Final Version

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Disclaimer

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Editorial comment:

Contributions from ENCO, IRR and VCE are integrated in the main body of this report; as well as their comments to the two drafts which have been circulated.

In doing so, figures and attachment of the ENCO-contributions have been omitted, and the IRR-contribution has been significantly shortened and restructured.

The full contributions of ENCO and IRR, without any omissions and shortenings, are added to this report as Appendices 2 (IRR) and 3 (ENCO). Appendix 2 also includes a separate section with inputs for Item 6 (site seismicity).

In Section 6 of this report (Control Rods and Fuel Degradation), the results of the internal meeting on November 15, 2006 and the comments on the paper with the results of this internal meeting, which was circulated, are integrated.

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SUMMARY

The workshop with walkdown in Temelín was organized by ČEZ, the operator of Temelín NPP, and the Czech licensing authority SÚJB. There were two main purposes to this workshop:

1. To provide information on some of the safety issues (Items) which had been indicated by the Austrian side for further monitoring according to the Conclusions of the Melk process and the results of the technical projects performed subsequently in the years 2002 to 2005, and summarized in the Summary Monitoring Report (SMR) of June 2005 (five Items).
2. To provide information concerning a number of new safety issues (of varying significance) which have become manifest at Temelín NPP in the last months (five Items).

For each Item, a brief evaluation as well as a presentation of issues of further interest is provided in this summary.

At the end of the summary, the Items with high priority for follow-up (taking into account importance and urgency) are identified.

28.8 m Level and Related Topics

Evaluation

Vibrations: The current situation would require a very detailed regime of in-service inspections. However, the analyses under way are likely to achieve the results needed for an appropriate selection of countermeasures.

Water Hammer: The changes in the bubliks[1] could solve this problem. New analyses with improved modelling will provide better insights.

Application of SUPERPIPE concept: Deviations from the basic requirements (see SMR, 2.1.4) remain.

Plans for power uprate: A concise set of licensing calculations is required for the uprate.

Risk-Informed In-Service-Inspection: This concept should not be regarded as a patent remedy to reduce the tremendous amount of non-destructive testing required.

All other areas: SMR section 2.1.4. still appears to be valid.

Issues of Further Interest:

The following issues should be further monitored:

- The current work on the bubliks.
- The vibration limitation attempts at the High Energy lines.
- The further development of non-destructive testing applications; particularly in the context of Risk-Informed In-Service-Inspection.

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[1] Twofold pipe sections leading from the main steam line to the two entry nozzles of the steam-relief valves, forming a doughnut (in Czech: bublik) shaped piping arrangement linked to the steam line via a double T-joint.

Regarding other areas identified earlier, SMR section 2.1.4 still appears to be valid. All those areas therefore should be further observed in the future.

Reactor Pressure Vessel

Evaluation

Preliminary irradiation results: A possible fluence rate effect of the test reactor irradiation or incorrect irradiation temperatures could have diminished the embrittlement of the specimen, compared to the real reactor pressure vessel material.

Temelín irradiation capsules: The first capsules were reported withdrawn in May 2004. Evaluation was to take about one year. Nevertheless, no data were provided at the workshop. They were reported to be available in late 2006/early 2007.

Other key areas: SMR section 2.3.4 still appears to be valid.

Issues of Further Interest:

The following issues should be further monitored:

- Development of the embrittlement, with detailed information.
- A number further issues relevant for reactor pressure vessel integrity is identified in the SMR (2.3.4).

Integrity of Primary Loop Components

Evaluation

Under-cladding cracks: The question of the non-destructive testing capabilities to detect small under-cladding cracks (SMR 2.4) still remains open.

Test defects: It is doubtful whether test defects used for qualification of weld inspections correspond to the worst case (as already pointed out in SMR, 2.4).

Other key areas: Regarding other aspects of the quality of in-service inspection of main primary loop components, the SMR (2.4) still appears to be valid.

Issues of Further Interest:

The following issues should be further monitored:

- Capabilities of non-destructive testing for the detection of small under-clad cracks and their differentiation from cracks within the reactor pressure vessel cladding.
- Test defects used for qualification of weld inspections in the primary circuit – particularly regarding the use of worst-case test defects.
- Reactor pressure vessel inspection experience.

Severe Accidents Related Issues

Evaluation

Progress of safety upgrading: The information provided generally shows that the work on safety upgrading of the plant is being continued. In some areas the progress is significant.

Organization of Severe Accident Management: The corresponding issues have been generally solved. There is progress in the development of the technical measures needed, and the implementation is under way. The process is not finished yet and deserves further monitoring.

Issues of Further Interest:

The completion of severe accident analyses, expected in 2007, should be further monitored. Also, the further development regarding the design of technical measures deserves monitoring. In particular, this concerns the following points:

- Upgrading of hydrogen recombiners.
- Measures for enlargement of the molten core area.
- Stuffing of ex-core ionization chambers' channels (2007).
- Enlargement of coolant inventory inside containment.

The final, implemented solution should be thoroughly reviewed and verified against calculations.

Seismic Issues

Evaluation

New monitoring system: This is a clear improvement of the situation. Still open is the evaluation of the obtained data in order to come to a realistic assessment of the seismic hazard for the Temelín NPP site.

Issues of Further Interest:

According to the agenda, only one specific topic has been touched in detail in this meeting, leaving all the other topics open for further clarification (see SMR 2.6 and 2.7).

- It should be checked whether the Austrian recommendations presented at the workshop have been adopted and implemented in the probabilistic seismic hazard assessment planned as part of the 10 year safety review.
- It would be of interest to receive raw data of the monitoring system in order to carry out an assessment.

Control Rods and Fuel Degradation

Evaluation

Development of Issue: Control rod insertion reliability has been identified as safety issue for WWER-1000/320 NPPs in the early 1990s, by IAEA. It was expected that the problem would not be experienced at Temelín with the new Westinghouse core design. Nevertheless, difficulties did occur in the last years, particularly in Unit 1.

A growing number of rod control cluster assemblies failed to achieve full touch-down in the bottom position at tests. Furthermore, at a test on June 02, 2006, two cluster assemblies stopped above the hydraulic dampers and thus failed to meet the Limit Conditions (which have to be fulfilled at all times during operation).

Counter-Measures: Actions are taken which are based on the experiences in other plants with similar problems. It can be expected that progress will be made.

Issues of Further Interest:

Due to the potentially high relevance of this topic, there are a number of questions which should be followed up with urgency.

A part of these questions relate to the five requirements listed in the IAEA's "Issue Book" on WWER-1000/320s (1996), concerning:

- Operational counter-measures (operation at reduced power)
- Drop times and drop tests
- Fuel loading and burn-up strategies
- Tests of lifting and lowering forces
- Structural counter-measures (readjustment of upper internal core structure)

Other questions considered as relevant go beyond the IAEA requirements:

- Investigations concerning safe shutdown in accident situations
- Fuel replacement strategies (from Westinghouse to TVEL fuel)
- State of knowledge regarding root causes
- Consequences of test loads for components' lifetimes

Leaks from Fuel Rods:

Evaluation

General assessment of problem: There is no significant deviation from problems as commonly encountered with nuclear fuel cladding failures in many nuclear power plants.

Causes of problem: The extended in-service period of the Westinghouse fuel could be part of the problem, but also fretting of the grid-spacers.

Counter-measures: Removal of the failed fuel rods and the replacement by solid stainless steel rods appears to be the practice to deal with the problem. This is an adequate procedure.

Issues of Further Interest:

There is no immediate safety concern arising from this Item.

- The number of leaks, however, gives reason to further observe the development of the leakage rates.
- The fuel is supposed to be changed. This should reduce the failure rate and should be observed.

Reactor Pressure Vessel Head Degradation

Evaluation

Implications for leak before break: Leak before break might not be fulfilled, due to inadequate leakage detectability, at least for reactor pressure vessel leaks.

Counter-measures: The cleaning procedures performed were effective; non-destructive testing analyses have been performed. Aside from a check on feasible improvements of detection, the introduction of administrative action to help avoiding excessive leakages and consecutive corrosion is expected.

Issues of Further Interest:

The following issues should be further monitored:

- Improvements of leak-detection capability in the primary circuit.
- Administrative measures which will be introduced to help avoiding excessive spills and leakages.

Emergency Diesel Generators/Relevance of Forsmark Event for Temelín NPP

Evaluation

Consequences of the Forsmark event: According to the present state of knowledge on this event, there are no indications of any danger of a similar scenario occurring in Temelín. As the circumstances of the Forsmark event have not been fully clarified yet, however, the analysis of Temelín experts so far cannot be regarded as final.

Issues of Further Interest:

- The continuing work of Czech experts regarding the relevance of the Forsmark event for Temelín should be further monitored.

Broken Pre-Stressing Cable:

Evaluation

Incident at Temelín: One pre-stressing cable of the containment structure has been found broken in Temelín at the occasion of a test.

Generic nature of problem: The problem of pre-stressing cable failure seems to be generic for pre-stressed concrete containments, and therefore also for WWER-1000s. There is consensus that the Russian system is not well suited to the test procedures. New monitoring technologies have been developed to check cables without destructive liftoff testing. These technologies should also be applied to nuclear plants.

Issues of Further Interest:

- The cause for the breaks should be found out on a generic level. After it has been found, the test procedures have to be adapted accordingly.
- With respect to the reported broken cable there are a number of open questions (see section 10 of this report for details), which should be followed by the Austrian side. The question whether this is a systematic problem is of main interest and has to be answered, as similar incidents have been reported from other plants.

General Impressions from Walkdown and Workshop

The program of the whole event was well balanced, with the first day reserved for the walkdown in the plant itself and the second day used for presentations of Czech experts and discussion of points of concern.

The walkdown provided a unique chance to get an overall impression of the plant and its condition. Even critical questions could be discussed. The open spirit of the operator's presentation and explanation shall be particularly mentioned.

The walkdown has made possible a multitude of detailed observations which have been taken into account in the various sections of the report.

Regarding the presentations, the operators of Temelín NPP have presented analyses demonstrating that the plant carefully follows all safety concerns connected with the operational experience in the plant and draws appropriate conclusions. Temelín experts follow also related events in other NPPs. This has been demonstrated by timely in-depth analysis of recent event in Forsmark, which showed Temelín design to be robust and resistant to such hazards.

A significant amount of information has been provided and many questions were discussed. It might be advisable, however, to avoid too broad an agenda at future occasions and to focus the program on a smaller number of issues which could then be treated more in-depth.

Priorities for Follow-Up

The following Items appear to be the most important ones for follow-up, according to their safety significance and urgency.

Control Rods and Fuel Degradation (New Item)

This is a new problem. Control rods are parts of one of the most crucial safety systems of an NPP. There are a number of questions which should be followed up with urgency; measures are planned but also need to be followed up.

High priority, high urgency.

28,8 m Level and Related Topics (Roadmap Item 1)

Still unresolved safety issues persist (application of SUPERPIPE, water hammer impact, break locations… see SMR 2.1.4), at the same time there are acute vibration problems requiring modifications, against the background of an envisaged power uprate.

High priority, high urgency.

Reactor Pressure Vessel and Primary Circuit (Roadmap Items 3 and 4)

So far, no surveillance results have been provided; the first results should become available in the near future. Other issues are still open as well (pressurized thermal shock analyses, application of VERLIFE, detection of small under-cladding cracks… see SMR 2.3.4)

High priority, medium to high urgency (embrittlement progresses only gradually).

ZUSAMMENFASSUNG

Der Workshop mit Anlagenbegehung wurde von ČEZ, der Betreiberin des KKW Temelín, und der Tschechischen Genehmigungsbehörde SÚJB organisiert. Der Workshop hatte hauptsächlich zwei Aufgaben:

1. Die Übermittlung von Informationen über einige der sicherheitsrelevanten Fragestellungen (Punkte), die von der österreichischen Seite für weiteres Monitoring ausgewiesen worden waren – entsprechend den Schlussfolgerungen aus dem Melker Prozess sowie den Ergebnissen der technischen Projekte, die danach in den Jahren 2002 bis 2005 durchgeführt und in einem Abschlussbericht (Summary Monitoring Report, SMR) vom Juni 2005 zusammengefasst wurden (fünf Punkte).
2. Die Übermittlung von Informationen über eine Anzahl von neuen sicherheitsrelevanten Themen (von unterschiedlicher Bedeutung), die in den letzten Monaten im KKW Temelín aktuell geworden sind (fünf Punkte).

Für jeden dieser Punkte wird in dieser Zusammenfassung eine kurze Bewertung gegeben, sowie eine Darstellung jener Fragen, die von weiterem Interesse sind.

Am Ende der Zusammenfassung werden die Punkte mit hoher Priorität für weitere Arbeiten (unter Berücksichtigung von Bedeutung und Dringlichkeit) identifiziert.

28,8 m Bühne und dazugehörige Themen:

Bewertung

Vibrationen: Die derzeitige Situation würde ein sehr detailliertes Regime von Wiederkehrenden Prüfungen erfordern. Die Analysen, die gerade durchgeführt werden, werden jedoch wahrscheinlich jene Ergebnisse liefern, die für eine angemessene Auswahl von Gegenmaßnahmen benötigt werden.

Wasserschlag: Die Änderungen bei den sogen. Bubliks[2] könnten dieses Problem lösen. Neue Analysen mit verbesserter Modellierung werden bessere Erkenntnisse liefern.

Anwendung des SUPERPIPE Konzeptes: Es verbleiben Abweichungen von den grundlegenden Anforderungen (siehe SMR, 2.1.4).

Pläne für Leistungssteigerungen: Für die Leistungssteigerungen wird ein präzise definierter Satz von Berechnungen als Basis für die Genehmigung benötigt.

Risiko-informierte Wiederkehrende Prüfungen: Dieses Konzept sollte nicht als Patentlösung betrachtet werden, um den enormen Unfang der erforderlichen Wiederkehrenden Prüfungen zu reduzieren.

Alle anderen Bereiche: Abschnitt 2.1.4 des SMR erscheint nach wie vor als gültig.

[2] Zweifache Rohrabschnitte, die von der Frischdampfleitung zu den beiden Eingangsstutzen der Dampf-Entlastungsventile führen – in Form eines Torus, der mit einer doppelten T-Verbindung an der Dampfleitung hängt.

Fragen von weiter gehendem Interesse:

Folgende Fragen sollten weiter verfolgt werden:

- Die laufenden Arbeiten an den Bubliks.
- Die Versuche, die Vibrationen der hochenergetischen Rohrleitungen zu begrenzen.
- Die weitere Entwicklung der Anwendung von Zerstörungsfreien Prüfungen; insbesondere im Zusammenhang mit Risiko-informierten Wiederkehrenden Prüfungen.

In anderen Bereichen, die früher identifiziert wurden, scheinen die Aussagen in Abschnitt 2.1.4 des SMR nach wie vor gültig. Alle diese Bereiche sollten daher in Zukunft weiter beobachtet werden.

Reaktordruckbehälter

Bewertung

Vorläufige Bestrahlungs-Ergebnisse: Ein möglicher Flussdichte-Effekt bei der Bestrahlung im Testreaktor, oder eine falsche Bestrahlungstemperatur könnten die Versprödung der Proben verringert haben, verglichen mit dem tatsächlichen Material des Reaktordruckbehälters.

Voreilproben in Temelín: Es wurde berichtet, dass die ersten Kapseln im Mai 2004 entnommen worden sind. Die Auswertung sollte etwa ein Jahr dauern. Dennoch wurden auf dem Workshop keine Daten präsentiert. Es wurde berichtet, dass sie Ende 2006/Anfang 2007 verfügbar sein sollten.

Andere wichtige Bereiche: Abschnitt 2.3.4 des SMR erscheint nach wie vor als gültig.

Fragen von weiter gehendem Interesse:

Folgende Fragen sollten weiter verfolgt werden:

- Entwicklung der Versprödung, mit detaillierter Information.
- Eine Anzahl weiterer Themen, die für die Integrität des Reaktordruckbehälters von Bedeutung sind und bereits im SMR (2.3.4) identifiziert wurden.

Integrität der Primärkreislaufkomponenten

Bewertung

Unterplattierungs-Risse: Die Frage nach der Leistungsfähigkeit der Zerstörungsfreien Prüfungen beim Nachweis von Unterplattierungs-Rissen (SMR 2.4) ist nach wie vor offen.

Test-Fehler: Es bestehen Zweifel ob die Test-Fehler, die für die Qualifikation der Prüfungen der Schweißnähte benützt wurden, tatsächlich dem schlimmsten Fall entsprechen. (Darauf wurde bereits im SMR, Abschnitt 2.4, hingewiesen.)

Andere wichtige Bereiche: Im Hinblick auf andere Aspekte der Qualität Wiederkehrender Prüfungen der wichtigsten Komponenten des Primärkreislaufes erscheinen die Aussagen im SMR (2.4) als nach wie vor gültig.

Fragen von weiter gehendem Interesse:

Folgende Fragen sollten weiter verfolgt werden:

- Die Leitungsfähigkeit der Zerstörungsfreien Prüfungen beim Nachweis kleiner Risse unter der Plattierung, und ihrer Unterscheidung von Rissen innerhalb der Plattierung des Reaktordruckbehälters.
- Die Auswahl der Test-Fehler, die für die Qualifikation der Prüfungen der Schweißnähte benützt werden – insbesondere im Hinblick auf die Anwendung von Test-Fehlern, die dem schlimmsten Fall entsprechen.
- Erfahrungen bei der Prüfung des Reaktordruckbehälters.

Fragen im Zusammenhang mit Schweren Unfällen

Bewertung

Fortschritt bei der sicherheitstechnischen Ertüchtigung: Die vorgelegten Informationen zeigen insgesamt, dass die Arbeiten zur Nachrüstung der Anlage fortgesetzt werden. In manchen Bereichen sind die Fortschritte bedeutend.

Organisation des Programms für den internen Notfallschutz: Die in diesem Zusammenhang bestehenden Probleme wurden überwiegend gelöst. Es gibt Fortschritte bei der Entwicklung der erforderlichen technischen Maßnahmen, und ihre Umsetzung ist im Gange. Der Prozess ist allerdings noch nicht beendet und sollte weiter verfolgt werden.

Fragen von weiter gehendem Interesse:

Der Abschluss der Analysen schwerer Unfälle, der 2007 erwartet wird, sollte weiter verfolgt werden. Ebenso verdienen die weiteren Entwicklungen im Zusammenhang mit der Ausführung technischer Maßnahmen eine Beobachtung. Insbesondere betrifft dies die folgenden Punkte:

- Ertüchtigung der Wasserstoff-Rekombinatoren.
- Maßnahmen zur Vergrößerung der Fläche für den geschmolzenen Reaktorkern.
- Verstopfen der Kanäle für die Ionisationskammern außerhalb des Kerns (2007).
- Vergrößerung des Kühlmittel-Inventars innerhalb des Containments.

Die endgültigen, umgesetzten Lösungen sollten gründlich bewertet, und durch Berechnungen verifiziert werden.

Seismische Fragen

Bewertung

Neues Überwachungssystem: Dies stellt eine deutliche Verbesserung der Situation dar. Offen ist noch die Auswertung der erfassten Daten, um zu einer realistischen Bewertung der seismischen Gefährdung am Standort Temelín zu gelangen.

Fragen von weiter gehendem Interesse:

Entsprechend der Tagesordnung wurde lediglich ein spezielles Thema beim Workshop im Detail behandelt. Alle anderen Fragen bleiben einer künftigen Klärung vorbehalten (siehe SMR 2.6 und 2.7).

- Es sollte überprüft werden, ob für die probabilistische seismische Gefahrenabschätzung, die als Teil der 10-Jahres-Sicherheitsüberprüfung geplant ist, die auf dem Workshop vorgetragenen österreichischen Empfehlungen angenommen und umgesetzt wurden.
- Es wäre von Interesse, Rohdaten vom Überwachungssystem zu erhalten, um eine Bewertung durchzuführen.

Kontrollstäbe und Schädigung des Brennstoffes:

Bewertung

Entwicklung des Problems: Die Zuverlässigkeit des Einfallens der Steuerstäbe in den Kern wurde für Kernkraftwerke vom Typ WWER-1000/320 in den frühen 90er Jahren von der IAEA als Sicherheitsproblem (safety issue) identifiziert. Es wurde erwartet, dass dieses Problem in Temelín aufgrund der neuen Kernauslegung durch Westinghouse nicht auftritt. Dennoch kam es in den letzten Jahren zu Schwierigkeiten, insbesondere in Block 1. Eine wachsende Zahl von Steuer-Elementen (rod control cluster assemblies) erreichte bei Tests nicht die tiefste Position. Darüber hinaus stoppten bei einem Test am 02. Juni 2006 zwei Steuer-Elemente in einer Position über dem hydraulischen Dämpfer und genügten damit nicht den „Limit Conditions“ (d. s. Bedingungen, die während des Betriebes jederzeit eingehalten werden müssen).

Gegenmaßnahmen: Es werden Maßnahmen ergriffen, die auf den Erfahrungen in anderen Anlagen mit ähnlichen Problemen beruhen. Es ist zu erwarten, dass es Fortschritte geben wird.

Fragen von weiter gehendem Interesse:

Aufgrund der großen, potenziellen Bedeutung dieses Punktes gibt es eine Reihe von Fragen, die mit Dringlichkeit weiter verfolgt werden sollten.

Ein Teil dieser Fragen bezieht sich auf die fünf Anforderungen, die im „Issue Book“ der IAEA für Kernkraftwerke vom Typ WWER-1000/320 (1996) aufgelistet sind:

- Betriebliche Gegenmaßnahmen (Betrieb bei verringerter Leistung)
- Fallzeiten und Falltests
- Brennstoff-Beladungs- und Abbrand-Strategien
- Tests der Hebe- und Senkungskräfte
- Strukturelle Gegenmaßnahmen (Umjustierung der internen oberen Kernstruktur)

Andere Fragen, die als wichtig angesehen werden, gehen thematisch über die Anforderungen der IAEA hinaus:

- Untersuchungen betreffend sicheres Abschalten in Unfallsituationen
- Strategien zum Ersetzen des Brennstoffes (von Westinghouse- zu TVEL-Brennstoff)
- Wissensstand über die dem Probleme zugrunde liegenden Ursachen
- Konsequenzen der bei den Tests auftretenden Belastungen für die Lebensdauer von Komponenten

Leckagen aus Brennstäben

Bewertung

Allgemeine Einschätzung des Problems: Es gibt keine bedeutsame Abweichung von den Problemen, wie sie allgemein im Zusammenhang mit dem Versagen von Brennstoffhüllen in vielen Kernkraftwerken auftreten.

Ursachen des Problems: Die verlängerte Standzeit des Brennstoffs von Westinghouse könnte zu dem Problem beitragen, ebenso wie Abnutzung durch Reibung an den Abstandhaltern.

Gegenmaßnahmen: Das Entfernen der beschädigten Brennstäbe und ihr Ersetzen durch solide Stäbe aus Edelstahl scheint die Praxis zu sein, mit der dieses Problem behandelt wird. Dies ist ein angemessenes Vorgehen.

Fragen von weiter gehendem Interesse:

Dieses Problem gibt keinen Anlass für kurzfristige Besorgnis aus Sicherheitsgründen.

- Die Anzahl der Leckagen legt es allerdings nahe, die Entwicklung der Leckage-Raten weiter zu beobachten.
- Es ist davon auszugehen, dass der Brennstoff ausgetauscht wird. Dies sollte die Versagensrate reduzieren und sollte weiter beobachtet werden.

Schäden am Dyeckel des Reaktordruckbehälters

Bewertung

Bedeutung für die Anwendung von „Leck-vor-Bruch“: Die Voraussetzungen für die Annahme von Leck-vor-Bruch sind möglicherweise nicht erfüllt, aufgrund von nicht ausreichender Detektierbarkeit von Leckagen. Dies könnte insbesondere für Lecks am Reaktordruckbehälter gelten.

Gegenmaßnahmen: Die angewandten Reinigungsverfahren waren effektiv. Untersuchungen mit zerstörungsfreien Prüfungen wurden durchgeführt. Neben einer Überprüfung, welche Verbesserungen beim Nachweis von Leckagen machbar sind, wird die Einführung administrativer Vorgehensweisen erwartet, die helfen sollen, übermäßige Leckagen und darauf folgende Korrosion zu vermeiden.

Fragen von weiter gehendem Interesse:

Folgende Fragen sollten weiter verfolgt werden:

- Verbesserung der Leistungsfähigkeit der Leckage-Erkennung im Primärkreislauf.
- Administrative Maßnahmen, die eingeführt werden, um übermäßiges Verschütten und übermäßige Leckagen vermeiden zu helfen.

Notstromdieysel/Bedeutung des Ereignisses in Forsmark für das KKW Temelín

Bewertung

Konsequenzen des Forsmark-Ereignisses: Nach dem derzeitigen Wissensstand über dieses Ereignis gibt es keine Anzeichen dafür, dass eine ähnliches Szenario in Temelín eintreten könnte. Da die näheren Umstände des Forsmark-Ereignisses noch nicht vollständig geklärt wurden, kann die Analyse der Experten von Temelín allerdings noch nicht als endgültig angesehen werden.

Fragen von weiter gehendem Interesse:

- Die fortdauernden Untersuchungen der Relevanz des Forsmark-Ereignisses für Temelín durch die tschechischen Experten sollten weiter verfolgt werden.

Gerissenes Spannseil

Bewertung

Vorfall in Temelín: Ein Spannseil der Containment-Struktur in Temelín wurde bei einem Test gerissen vorgefunden. Dieses Thema wurde von der tschechischen Seite nicht ausreichend erläutert.

Generischer Charakter des Problems: Das Problem des Versagens von Spannseilen im Beton scheint für Containments aus Spannbeton generisch zu sein, und damit auch für WWER-1000-Anlagen. Es besteht Konsens, dass das russische System für die Testverfahren nicht gut geeignet ist. Neue Technologien zur Überwachung wurden entwickelt, um Seile ohne zerstörendes, abhebendes Testen zu überprüfen. Diese Verfahren sollten auch bei Atomanlagen angewandt werden.

Fragen von weiter gehendem Interesse:

- Die Ursache für das Reißen des Seils sollte auf allgemeiner, generischer Ebene gefunden werden. Nachdem sie identifiziert wurde, müssen die Testverfahren entsprechend adaptiert werden.
- Im Hinblick auf den Bericht über das abgerissene Spannseil gibt es mehrere offene Fragen (zu den Details siehe Abschnitt 10 dieses Berichtes), die von der österreichischen Seite weiter verfolgt werden sollten. Am wichtigsten ist die Frage, ob dies ein systematisches Problem ist. Diese Frage muss geklärt werden, angesichts ähnlicher Vorfälle, die aus anderen Anlagen berichtet wurden.

Allgemeine Eindrücke von der Anlagenbegehung und dem Workshop

Das Programm der Gesamtveranstaltung war gut ausgewogen – der erste Tag war für die Begehung der Anlage reserviert, während der zweite Tag für Präsentationen durch tschechische Experten und Diskussion wichtiger Punkte genützt wurde.

Die Begehung bot eine einmalige Chance, einen Gesamteindruck von der Anlage und ihrem Zustand zu erhalten. Auch kritische Fragen konnten diskutiert werden. Der unvoreingenommene Geist der Darstellungen und Erklärungen der Betreiber verdient es, besonders erwähnt zu werden.

Die Begehung mache eine Vielzahl von Beobachtungen möglich, die in den verschiedenen Abschnitten dieses Berichtes berücksichtigt wurden.

Bei den Präsentationen haben die Betreiber von Temelín Analysen vorgelegt, die zeigen, dass allen Sicherheitsbedenken, die sich aus der Betriebserfahrung ergeben, sorgfältig nachgegangen wird, und dass daraus angemessene Schlüsse gezogen werden. Die Experten in Temelín verfolgen auch relevante Ereignisse in anderen Kernkraftwerken. Die zügig durchgeführte, gründliche Analyse des kürzlich in Forsmark eingetretenen Ereignisses hat dies bestätigt. Diese Analyse zeigte im Übrigen, dass die Auslegung in Temelín gegenüber derartigen Gefahren robust und widerstandsfähig ist.

Es wurde insgesamt viel an Informationen zur Verfügung gestellt; zahlreiche Fragen wurden diskutiert. Es könnte allerdings ratsam sein, in Zukunft eine zu breite Tagesordnung zu vermeiden und das Programm auf eine kleinere Zahl von Themen zu beschränken, die dann in größerem Detail behandelt werden könnten.

Prioritäten für weiterführende Arbeiten

Folgende Punkte erscheinen im Hinblick auf ihre Bedeutung für die Sicherheit sowie ihre Dringlichkeit als die wichtigsten für weiterführende Arbeiten:

Kontrollstäbe und Schädigung des Brennstoffes (Neuer Punkte)

Hier handelt es sich um eine neues Problem. Die Kontrollstäbe gehören zu einem der wichtigsten Sicherheitssysteme eines Kernkraftwerkes. Es gibt eine Reihe von Fragen, die dringend weiter verfolgt werden sollten. Maßnahmen sind geplant, sollten jedoch ebenfalls verfolgt werden.

Hohe Priorität, hohe Dringlichkeit.

28,8 m Bühne und dazugehörige Themen (Roadmap Punkt 1)

Zu diesem Punkt bestehen nach wie vor ungelöste Sicherheitsfragen (Anwendung des SUPERPIPE-Konzeptes, Wirkung eines Wasserschlages, Orte von Brüchen… siehe SMR 2.1.4). Gleichzeitig gibt es akute Probleme mit Vibrationen, die Modifikationen erfordern – vor dem Hintergrund einer geplanten Leistungserhöhung.

Hohe Priorität, hohe Dringlichkeit.

Reaktordruckbehälter und Primärkreislauf (Roadmap Punkte 3 und 4)

Bisher wurden keine Ergebnisse von Voreilproben zur Verfügung gestellt; die ersten Ergebnisse sollten in naher Zukunft verfügbar werden. Andere Fragen sind ebenfalls immer noch offen (Analysen von Thermoschock unter hohem Druck, Anwendung von VERLIFE, Nachweis kleiner Risse unter der Plattierung… siehe SMR 2.3.4).

Hohe Priorität, mittlere bis hohe Dringlichkeit (die Versprödung schreitet nur allmählich fort).

INTRODUCTION

The workshop with walkdown in Temelín was organized by ČEZ, the operator of Temelín NPP, and the Czech licensing authority SÚJB. There were two main purposes to this workshop:

1. To provide information on several safety issues (Items) which had been indicated by the Austrian side for monitoring on a technical level according to the Conclusions of the Melk process. The aim of monitoring is to get information about the level of completeness and the appropriateness of the solutions being applied by Temelín NPP to resolve all safety issues, as indicated within the framework of the Road Map for Implementation of Annex I and Annex II of the Conclusions of the Melk Process and Follow-up, established in the Czech-Austrian Bilateral Agreement in November 2001. For each Item relevant to safety, a specific technical project had been initiated and performed in the years 2002 to 2005. In this framework, a number of issues could be clarified; and some issues for the future exchange of information were identified. Some of those have been treated at the workshop.
2. To provide information concerning a number of new safety issues (of varying significance) which have become manifest at Temelín NPP in the last months and which have, to some extent, raised concerns among the Austrian experts as well as the public in Austria.

More information about the monitoring on a technical level according to the ETE Road Map 2002 – 2005 can be found in the Final Monitoring Reports of the projects, and in the Summary Monitoring Report of June 2005, all published by Umweltbundesamt, Vienna.

The agenda of the workshop as well as a listing of the presentations provided by the Czech side can be found in Appendix 1.

For each of the Items covered at the workshop, one of three Austrian technical support organizations (partly with sub-contractors) was responsible:

- ENCONET Ges.m.b.H., Vienna (ENCO)
- Institute of Risk Research, University of Vienna (IRR)
- VCE – Vienna Consulting Engineers Holding GmbH (VCE)

Each of those organizations has provided inputs for this report for the Items assigned to them; further information (detailed reports, presentation) from the TSOs is contained in Appendices 2 – 4 of this report. To some extent, organizations have also commented on Items which were the main responsibility of others, in their Appendix.

At the workshop, questions of valve qualification (Item 2; PN 3[3]) were not on the agenda and not dealt with; also no questions were raised in connection with turbine generator vibrations. Those two Items are briefly discussed in Appendix 2 to this report (detailed report of IRR).

The following Items which had already been subject to Road Map projects have been covered by the workshop:

- 28.8 m Level and Related Topics – High Energy Pipelines and Valves, Turbine (Items 1, 2; PN 2, 3) (IRR)
- Reactor Pressure Vessel (Item 3; PN 9) (IRR)

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[3] Item no. and corresponding project number. This terminology is presented in the SMR, which also contains a listing of all projects.

- Integrity of Primary Loop Components (Item 4; PN 10) (ENCO)
- Severe Accidents Related Issues (Item 7b; PN 7) (ENCO)
- Seismic Issues (Items 6 and “Seismic Design”; PN 6,8) (VCE)

The following new Items were on the workshop’s agenda:

- Control Rods and Fuel Degradation (IRR)
- Leaks from Fuel Rods (IRR)
- RPV Head Material Degradation (IRR)
- Emergency Diesel Generators – Relevance of Forsmark Event for Temelín NPP (ENCO)
- Broken Pre-stressing Cable (VCE)

The information provided at the workshop is summarized and evaluated for each Item; issues of further interest are identified. Finally, the attempt was made to identify the Items of highest priority for follow-up.

The Item “Control Rods and Fuel Degradation” (Section 6 of this report), which concerns current problems at Temelín in an important safety system, has been identified as particularly relevant after the workshop. Therefore, additional considerations were performed by Austrian experts regarding the questions to be followed up in connection to control rod insertion problems. Apart from ENCO and IRR, an expert from the German TSO Physikerbüro Bremen took part in those considerations.

The results of this additional work are summed up in an addendum to section 6.

The main body of the report represents the consensus of all experts who have contributed to writing and structuring the report. The content of the Appendices 2 – 4 remains the sole responsibility of the respective authors.

1 28.8 M LEVEL AND RELATED TOPICS – HIGH ENERGY PIPELINES AND VALVES, TURBINE (ITEMS 1, 2; PN 2, 3)

(Contributed by IRR)

Introduction

As mentioned in the Final Monitoring Report and the Summary Monitoring Report (June 2005), extensive work was accomplished by the plant operators and their technical support organisations to consolidate the safety case regarding the high energy pipelines. Improvements had been achieved in a number of areas.

An integrated approach of prevention, protection, qualification and mitigation measures (Defence-in-Depth), however, was followed only partially so far. For sections of the pipes, the safety case relied on break exclusion only. Reliable compensation of this circumstance according to western practice had not yet been demonstrated. A number of issues remained to be clarified.

At the workshop, questions of valve qualification (Item 2; PN 3) were not on the agenda and hence, not dealt with; neither were questions in connection with turbine generator vibrations.

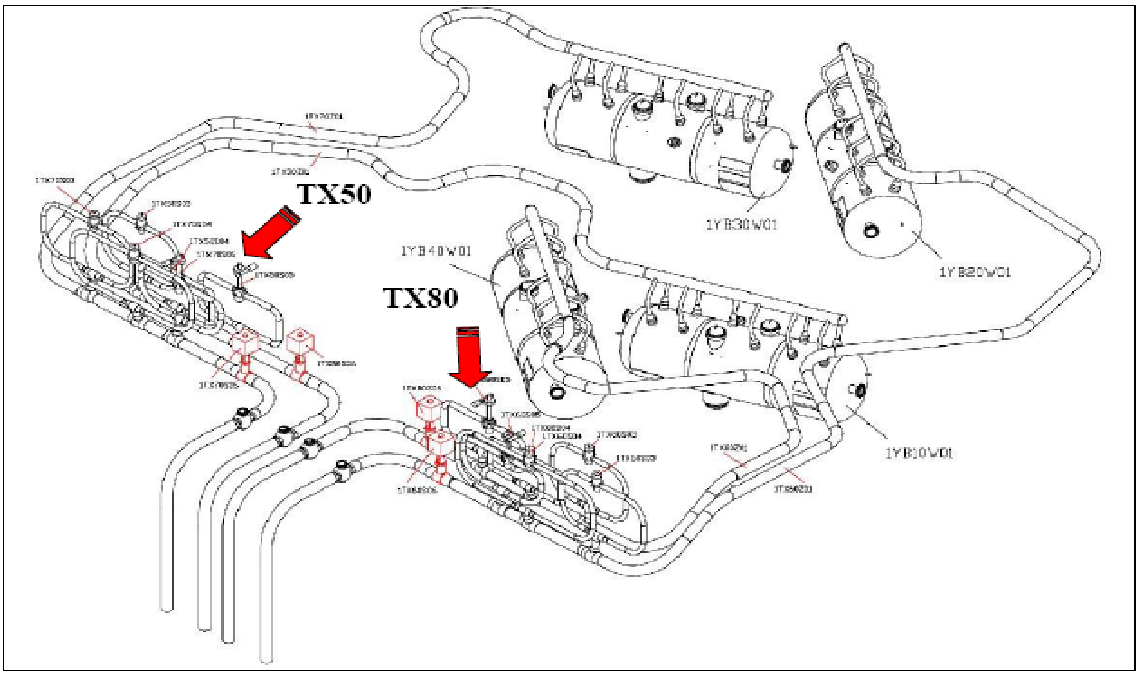


Fig. 1.1: Main steam lines with steam generators, bubliks and valves (presentation "Improvement in A820 Compartment", see Appendix 1 for full reference)

Summary of New Information Provided

The **SUPERPIPE Concept application** is not supposed to be extended to the bubliks' areas. (Bubliks are piping loops connected to the main steam lines, each with a T-joint and a T-valve.) Conceptual improvement measures and their implementation are likely to render the situation there more acceptable. The basic Czech approach was to declare the main secondary steam lines and the main secondary water lines to be no break zones, manufactured according to modified USNRC and French requirements and endorsed by the required analyses and stepped-up ISI. This approach was said to have been reviewed by NRC experts in the field. Their statement was that the physical lay-out could be accepted. The NRC review's results, however, were not discussed nor provided. They would have been helpful for a better understanding of the issue.

The bubliks remain a postulated break zone. Regarding the **selection of break locations**, the revised pipe support/suspension concept was applied to the bubliks with GERB dampers (supplied by GERB Vibration Control Systems, an international group of companies) at distinguished locations. The consideration to reduce maximum load combinations led to the limitation of motion, acceleration, frequencies and resonance of the pipe sections there. The investigations into the vibration properties of the bubliks indicate that it is difficult to demonstrate that breaks at the "critical" break locations will not be initiated by maximum load cases. This demonstration would require confirming that stresses and strains would remain within allowable limitations. Besides this, it must be demonstrated for the break events assumed to occur that they have negligible secondary effects in the area in question.

Verification has improved with respect to the adjacent bubliks; the loads' carry over resulting from mechanical and thermal energy discharges there have been finally tackled with apparently adequate means and efforts. Those results are expected to help to improve the situation.

Improvement of steam lines operation conditions in the A820 compartment as presented at the workshop was based on the work at IAM, Brno (a 100% subsidiary of the NRI Řež), related to the hydraulic and thermal load analyses of the bubliks' piping sections of the MSSL (Main Secondary Steam Line) at the 28,8-m-level including the MSSSVs (Main Secondary Steam Safety Valves) and the BRUA-relief-valves there.

The evaluation of the results from the Preoperational Tests revealed the necessity of an optimization of selected equipment (turbine, safety systems, high energy lines etc.). At the same time an investigation of the possibility for future utilization of margins for power uprating is under way.

The high energy lines' vibrations are near the upper limits of acceptance mostly applied at WWER plants (20 mm/s) at least for one of the main secondary steam lines of ETE (2TX50S05), with a speed of 18 mm/s. At present the situation is attributed to engine vibrations (vibrations resulting from valve discharge operation) and, more importantly, to excessively induced piping vibration.

Because of the stricter limit that would be imposed by ASME Code requirements, in this case amounting to 12,5 mm/s, there is a need of either avoiding the no break zone rules application (which was evidently envisaged originally), or to reduce the vibration at least to allowable limits.

The diagnosed parameters for the vibrations are pressure pulsations of maximal 50 kPa at the pipe section TX80Z01 with a frequency slot at 46 Hz. The power output range in question is 97 % unit power; the expected pressure oscillations are at 80 kPa resulting in a need for modifications according to the identified loading pattern. For the suspension of the main steam lines dampers on TX50 (TX60, TX70 do not show vibrations) have been installed (for 46 Hz, as prepared by VibroSeismcompany). The geometry of the piping system is to be redesigned.

A decision on the redesign and geometry modifications will be based on acoustic and piping modal analyses, on the analyses of acoustic wave lengths and excitation vibration analyses. Since the problem is known in Russian and Ukrainian plants, these were visited by IAM for further qualification (Balakovo, Wolgodonetsk and Chmelnitzki) of the strategies to be applied.

In the conclusions of IAM it was stressed that no flaws occurred and no damage was caused by the 46 Hz vibrations to the safety valves or to the piping. This situation is also known from the other WWER-1000 plants visited by the team of IAM investigating the Temelín issue.

Nevertheless, the unacceptably high velocity of the vibrations requires countermeasures in order to improve the current operational situation, and to resolve this problem also well in advance for an eventual power up-rating of the two blocks of Temelín NPP's.

Regarding **NDT**, the UT testblocks for the fillet welds were presented.

The Risk-Informed In-Service-Inspection (RI-ISI) concept will, according to NRI Řež, only be used to define the ISI priorities, not for the reduction of the NDT effort. All circumferential welds have been tested meanwhile. Only one more severe defect had been detected, apparently a lack of fusion from fabrication, which could be analysed with a SAFT-like approach (SAFT = synthetic aperture focusing technique) and was accepted by the authority.

Evaluation

In the **application of the SUPERPIPE concept**, there are deviations from the basic requirements according to the U.S. ASME code – regarding the restrictions on the pipe length, the no branching requirement, the design of the fixed points load carrying elements and the weldments there. For ASME Section III Code applications this would mean that the applicant must file a Code Case and have it resolved by the appropriate ASME Code Chapter experts committee. The decisions of the committee are then binding.

Concerning **recriticality** after multiple line breaks, the findings of the SMR (section 2.14) still appear valid, both for the old and the new core configurations.

It remains to be seen whether the changes envisaged and implemented at the bubliks make **water hammer impact** also a problem of the past. It should be recalled that previously (during the ETE monitoring process), modelling was based on a rather simplistic approach – modelling the pipe sections as linear/membrane elements, and the pressure pulses like superimposed pressure waves acting at the end of the pipes T-joints and internally.

This picture has changed dramatically with the newly introduced analyses for the steady flow cases modelled, in order to determine vibration induced by flow pulsations. It was mentioned that the redesign of the geometry will be done by coupling vortex shedding and acoustic resonance analyses.

The design modifications envisaged and those implemented already take into account the results of:

1. Acoustic modal analysis
2. Piping system modal analysis
3. Analysis of acoustic valves' length
4. Forced vibration analysis

The nature of the loads affecting the structures will become more visible this way, than via rough calculations seeking to pin down maximum expected material usage values at individual locations on the piping, which had to be selected "by hand".

In essence the models used now allow a limited cross check also of the proper choice of postulated break locations. This applies in case one can make sure the loads are combinations of all significant contributors and the way their description follows limiting events is sufficiently accurate also with regard to their history.

Generally seen, this seems to be a good start.

With respect to the **pipelines' fixed points** at the transition to the turbine hall the findings of the SMR (2.1.4) are still valid. The fixed points are still not able to resist the loads which can occur.

The stress at certain locations already reached permissible limits with the original load cases which were based on the nominal power to be generated at that point in the development of ETE. Therefore, the **aim to increase the power output** will require a concise set of licensing calculations. They should not be mere updates of the most recent assessment calculations used for the current license.

Consolidation of a comprehensive and largely covering assessment should be the prime aim of the safety assessment. The plant in its present state already represents a challenge in this respect. This is further aggravated when one takes into consideration the imminent changes to core, core control and the secondary side as well – also considering the turbo generator and the minor changes to the steam dumping options.

The current situation, with **vibrations** on one of the lines at the acceptable limits, would require a very detailed ISI regime to be put in action in order to record possible early damage to the piping, the joints and the valves. This is particularly important since the ASME code requirements are held up as providing the limitations on the steady state usage of the components.

Permanent vibration monitoring should be recommended in order to detect changes in the system long before a crack can be found by conventional NDT methods.

The analyses under way are likely to achieve the results needed for an appropriate selection of countermeasures and even combinations of suitable damping. It will have to be guaranteed, however, that the nodalization effects can be definitely kept apart from real sources of excitation and "vortex" flow, and the T-joints can be represented in their energy-carry-over properties well and stable enough over the simulation periods for the different operational modes of the main piping and the bubliks as well.

Fluid loads as well as pressure pulsations are part of the problem not only with the bubliks' layout, but also with the operability of safety and relief valves at the secondary side of the steam generators in normal operation and adverse conditions utilizing the steam setting off options.

Once the solution is adopted and implemented, there is a clear need for documented effects analyses.

The entire undertaking should never be seen as a “precautionary” measure. It is an operational fix to a problem that is rather commonly encountered after trial periods in large steam line circuits. Resolving the problem should not be delayed unduly. The solution of the vibrations problem is particularly important also in view of the power uprate envisaged.

The current work on the bubliks, once finalized, could close an important gap in the qualification of the piping connected to the main piping with the **no-break-zone postulate** as SUPERPIPE.

The vibration limitation attempts at the High Energy Lines are a substantial contribution to safety of the secondary circuit entering and exiting the containment. If successfully implemented the trustworthiness of the pressure relief system connected to the secondary coolant circuit will be increased.

The **progress in almost all other areas** seems to be very limited. There were no programs mentioned which would lead the experts to expect major changes in the Item 1 shortcomings as identified in the SMR (2.1.4).

In the following context, in particular, no changes appear to have taken place: Material properties like tensile strength properties, which are used to demonstrate the fulfillment of mechanical stress criteria, are neither the nominal properties according to the design rules, nor the minimum properties of the pipe material as guaranteed by the manufacturer. With either one of the latter, break exclusion criteria would not be met.

The material property values which are used are derived from specimen. Their identity with the actual pipe material has not yet been proven.

No new information on the access limitations for **NDT testblocks** was provided. An inspection had been carried out on the Friday before the workshop, but the results were not yet available.

RI-ISI should be applied with care. It should not be regarded as a patent remedy to reduce the tremendous amount of NDT at the 28,8 meter level!

The NDT inspections at the bubliks are not yet fully decided. The welds are included, the nozzle corners not yet. But there are considerations to include them.

Issues of Further Interest

The following issues should be further monitored:

- The current work on the bubliks.
- The vibration limitation attempts at the High Energy lines, including ISI results.
- The further development of NDT applications; particularly in the context of Risk-Informed Inspection.

As has been pointed out above, there seems to be little progress in other areas which were identified as requiring clarification in the SMR and the FMR. All those areas therefore should be further observed in the future. In particular, this should be seen in connection with the planned increase of the power output.

No specific preparatory work is required for the Austrian side to permit further discussion of these issues.

2 REACTOR PRESSURE VESSEL (ITEM 3; PN 9)

(Contributed by IRR)

Introduction

In the Final Monitoring Report and the Summary Monitoring Report (June 2005), it was stated that the Czech approach for pressurized thermal shock (PTS) analyses was in accordance with the state of science and technology, with respect to the concept, the methodology and the applied computer codes.

The surveillance program for the monitoring of embrittlement progression was found to represent a considerable improvement compared to other WWER-1000s.

The demonstration of reactor pressure vessel integrity has been conducted according to the VERLIFE method. In comparison with the IAEA "Guidelines on Pressurized Thermal Shock Analysis for WWER Nuclear Power Plants" (IAEA-EBP-WWER-08) of 1997, the VERLIFE method introduced a reduction of safety factors. Applying the 1997 IAEA Guidelines, a demonstration of the fulfilment of the requirements for reactor pressure vessel integrity would not have been possible for the whole lifetime of either unit.

In January 2006 a revised version of the IAEA-Guidelines (IAEA-EBP-WWER-08 (Rev.1)) has been published, which corresponds to the VERLIFE method's requirements. The introduction to this document states this explicitly:

"The revision of the report was performed in the frame of the IAEA Programme on Safety Analysis and Accident Management in the period November 2001 to March 2002. The revision of the guidelines was drafted by a group of experts during a meeting held 27–29 November, 2001. […] The document 'Unified Procedure for Lifetime Assessment of Components and Piping in WWER Nuclear Power Plants – VERLIFE' was prepared within the frame of the VERLIFE project of the EU 5th framework programme in the period 2001–2003. The subject of the VERLIFE procedure is much broader than that of these Guidelines. Several sections and appendices of VERLIFE document deal with integrity of RPV and PTS assessment. The preparation of the Revision 1 of the Guidelines was scheduled to facilitate harmonisation of both documents."

As a consequence of the retroactive introduction of reductions in the safety factors by the IAEA into the guidelines, and of other changes, the demonstration of integrity for the reactor pressure vessels of the Temelín NPP now corresponds to the IAEA prerequisites.

According to the revised guidelines, no safety factors have to be applied to stress intensity and shift in brittle fracture temperature in case of postulated accidents (in contrast to the earlier version). The criterion for the warm pre-stress (WPS) effect has been changed to 90 % of the load path maximum, as opposed to 80 % before. Regarding the selection of postulated defects' sizes, it is explicitly recommended to take the "high confidence of detection crack" as basis, with the application of a safety factor of 2. This point was not contained in the earlier version.

In the Temelín safety analyses, the embrittlement parameters which were used for the RPV materials correspond to those cited in the Russian Code. However, it has already been pointed out in the Final Monitoring Report of PN9 that Russian experts have questioned the conservatism of the values which were specified, based on experimental results.

On this basis – applying the VERLIFE methodology with which the IAEA guidelines are now harmonised, and using the embrittlement parameters of the Russian Code, reactor pressure vessel integrity could be demonstrated.

In the SMR (2.3.4), a number of additional important issues was identified for further treatment and attention, amongst them the surveillance program (initial results). At the workshop, regarding the reactor pressure vessel, the agenda was limited to this latter topic. Hence, other issues were not discussed.

The issue of PTS analyses and VERLIFE is treated in Appendix 2 to this report (detailed report of IRR).

Summary of New Information Provided

No specimen test data from the first set of irradiated samples are available so far. Therefore, no experimental validation of the projected progress of embrittlement is possible for the operator up to now. The Czech experts have provided a “preliminary” irradiation data forecast only.

The evaluation of the irradiated surveillance specimen withdrawn from the RPV in May 2004 (ETE-1) was announced to take place after one year (i.e. in summer 2005). Nevertheless, no results have been provided during this Workshop 2006. Also, no results were provided for the irradiated surveillance samples from ETE-2 which were withdrawn 2005. In September 2005 the surveillance samples were transported to the hot cells at Řež. The results of the mechanical testing were said to be available by late 2006/early 2007.

The next withdrawal of irradiation capsules is projected for 2008 (ETE-1) and 2009 (ETE-2).

The Czech presentation included “preliminary irradiation results” from material irradiated in a test reactor (fig. 2.1). The answer to questions of the Austrian experts’ team with respect to the materials irradiated within this program indicated that the irradiated samples were manufactured from acceptance test materials.

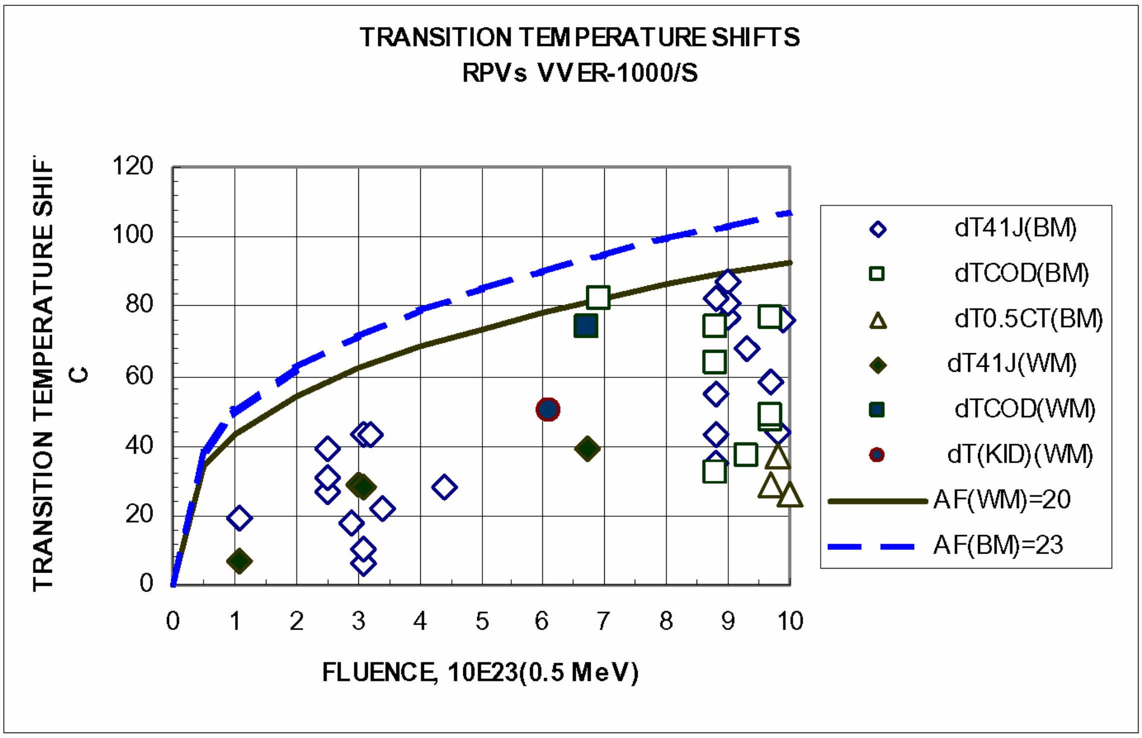


Fig. 2.1: Preliminary results of RPV Temelín 1 + 2 (presentation "Surveillance Programme", see Appendix 1 for full reference)

Evaluation

During the PN9-Workshop in May 2004, the Czech experts informed the Austrian experts' team that the first irradiation capsules had already been withdrawn in the same month, and the evaluation of the irradiated specimens would take about one year. Nevertheless, the expected data were not provided in September 2006. It was pointed out by the Czech side that the handling and evaluation of the samples constitutes a very complicated process. Details concerning the reasons for this delay were not provided.

The delay in evaluating the first set of specimen is potentially in conflict with the fact that the most rapid increase in embrittlement of the RPV materials is supposed to occur during the first years of irradiation (app. 5 years in general).

The Czech presentation on the "preliminary irradiation results" (see fig. 2.1) showed that all results of Charpy tests (which are a procedure to measure impact toughness) were below or close to the embrittlement curve as calculated according to the Russian standards. This means that a possible fluence rate effect due to the high lead factor (about 160) of the test reactor irradiation or incorrect irradiation temperatures within the test reactor could have diminished the embrittlement of the specimen in comparison with the material of the RPV itself.

The Austrian team of experts considers the information on the surveillance program results of significant importance for the experimental confirmation of the progress predicted for the embrittlement.

In accordance to the agenda, no other key areas of Item 3 were dealt with at the workshop. Hence, it appears that the shortcomings concerning this Item as identified in SMR (2.3.4) are still valid.

Issues of Further Interest

The Austrian team of experts recommends to gain access to information on the embrittlement development sufficiently detailed for evaluation. The notification about material properties as determined experimentally in the surveillance programme is of high importance in order to verify the compliance of material properties to the code specifications.

Information should also be sought on RPVI (reactor pressure vessel integrity) relevant issues such as core configuration, RPV internals and loads on those, main coolant line penetrations, vessel head and other penetrations, main flange tightness, coolant chemistry, hydrogen diffusion, corrosion, fatigue, surveillance measures ascertaining LBB applicability, actual RPVI verification and severe accident behaviour.

No specific preparatory work is required for the Austrian side to permit further discussion of these issues.

3 INTEGRITY OF PRIMARY LOOP COMPONENTS (ITEM 4; PN 10)

(Contributed by ENCO)

Introduction

In the Final Monitoring Report (May 2005), in-service-inspection (ISI) and non destructive testing (NDT) for the primary circuit in Temelín were found to be based on requirements as adopted by Western European countries and to be in most respects comparable to the western practices. Some open issues, however, remained.

Summary of New Information Provided

Quality of in-service inspection (ISI) of main primary loop components (RPV, coolant lines, primary side of SG, surge line) is an important aspect of safety of Temelín.

This issue was not on the workshop agenda, and was only touched occasionally in the context of other topics. Hence, with one exception, no special information was made available during the workshop in Temelín.

Since for many of the primary loop components made out of non cladded stainless steel, the qualification measures had not been fully completed at the time of the October 2004 workshop, and a request was established for information on the results of the ISI of the RPV, including the reports from ISI activities implemented in the meantime. So far, this request was not fulfilled.

In this context, Austrian experts have already expressed some doubts related to the use of non-worst-case test defects for the qualification of weld inspections on the primary coolant piping and the evaluation of actual inspection results (SMR 2.4).

Originally used methods for WWER-1000 were improved to correspond to the Western standards. Some issues related to applications remained.

This especially applies to the RPV inspection experiences and the role and efficiency of the TOFD (time-of-flight-diffraction) technique (ultrasound testing method with two angle probes). In accordance to the agenda, no reports or additional information were made available concerning this topic.

The Level of completeness and appropriateness of ISI/NDT for assuring integrity of primary loop components at Temelín, under all normal and **accidental conditions**, was identified as an important issue earlier (SMR 2.3.4, 2.4).

Although it has already been pointed out in the SMR, section 2.4, that the level of ISI and NDT at Temelín is comparable to Western European practice, this issue could not yet be completely resolved. It continues to be of interest and its resolution might depend on the availability of reports and documentation of a complete 4 years ISI inspection cycle on the primary circuit. This would establish a better basis to judge the quality of the ISI, than the reports about the diverse qualification used.

VERLIFE methodology requires 100% detectability of defects of certain depth. For Temelín, it is defined as half the PTS-relevant crack depth (10 mm under-clad).

Differentiation between under-clad cracks and cracks within RPV cladding is essential to assure applicability of the VERLIFE concept to the PTS. It has not been proven in trials, but time for improvement is available before reaching critical PTS at Temelín's RPVs.

The only information received at the workshop in Temelín regarding 100% detectability and differentiation between under-clad cracks and cracks within RPV cladding is a confirmation related to the assumptions relevant for defects that are in the vicinity of the under-cladding cracks: Those should be limited to the ferritic-base material. That means that the cladding is assumed to be intact at a location where NDT establishes indications for such a crack. Austrian experts were (verbally) informed (by a representative of NRI, Řež) during the September 2006 workshop that an additional NDT-technique using low-frequency eddy current inspection was implemented as a mandatory activity in cases when the UT inspection returns with an indication, in order to prove that there is an intact cladding where a UT indication in the near-cladding zone was observed. This was to be qualified during preparation of next ISI of RPV. Thus, the suggestion of the Austrian experts during the previous workshop was taken into account.

Evaluation

The quality of in-service inspection (ISI) of main primary loop components (RPV, coolant lines, primary side of SG, surge line) is an important aspect of safety of Temelín. This issue was not on the workshop agenda, and was only touched occasionally in the context of other topics. Hence, no additional written information (reports) concerning this issue was made available during the workshop in Temelín.

The most important remaining question in relation with Item 4/PN 10, Integrity of Primary Loop Components and NDT, as specified in the SMR (2.4), were the **"NDT-capabilities of detection of small under-cladding RPV cracks".** This question still remains open; some verbal information regarding eddy current testing only was given at the workshop.

This issue should be monitored and reconsidered on the basis of the reports and documentation of a complete 4 years inspection cycle on the primary circuit. This would establish a better basis to judge the quality of the ISI than the reports about the diverse qualification exercises that were originally presented.

In particular, it would be appreciated if the verbal information concerning differentiation of under-clad cracks and cracks within RPV cladding could be corroborated with actual reports documenting the inspection undertaken and the evaluation the results of those inspections.

Another issue requiring clarification concerns the test defects used for the qualification of weld inspections in the primary circuit and the evaluation of actual inspection results. Austrian experts expressed doubts in the FMR of the corresponding project as well as in the SMR (2.4) that the test defects used correspond to the worst cases; these doubts appear to be still valid.

Issues of Further Interest

The following issues should be further monitored:

- NDT-capabilities of detecting small under-clad cracks and differentiating them from cracks within the RPV cladding. Reports and documentation of a complete 4 year inspection cycle would provide the best basis.
- Test defects used for qualification of weld inspections in the primary circuit – particularly regarding the use of worst-case test defects.
- RPV inspection experience, in particular concerning the TOFD technique.

No specific preparatory work is required for the Austrian side to permit further discussion of these issues.

4 SEVERE ACCIDENTS RELATED ISSUES (ITEM 7B; PN 7)

(Contributed by ENCO)

Introduction

At the time of compilation of the Final Monitoring Report and the Summary Monitoring Report (June 2005), the development and implementation of the Temelín severe accident management (SAM) program had not been finalized. However, they were reported as well advanced. The overall concept and the approach to the implementation of the severe accident management guidelines (SAMG) packet were found to reflect good practice. The program is supported by severe accident analyses and a plant specific probabilistic safety assessment (PSA).

Some technical measures for severe accident management had not yet been introduced. Several issues had remained open and were identified in the Final Monitoring Report and the Summary Monitoring Report.

To provide further information, Severe Accident Related Issues were presented at the workshop by representatives of the operator in the paper “Temelín Accident Management Programme” (see Appendix 1 for full reference). They discussed:

- Completeness of analyses and resulting measures
- Organizational arrangements for SAMG not finalized at the time of the review
- Technical measures needed for prevention and mitigation of risk significant scenarios

Further information on this Item, particularly a number of figures illustrating the content of this section, can be found in Appendix 3 to this report (detailed report of ENCONET).

Some comments to this Item were also provided by IRR. They can be found in Appendix 2 to this report (detailed report by IRR).

Summary of New Information Provided

Analysis of containment phenomena

Analyses of containment phenomena were started in 2004 and full results will be available in 2007. In the analyses the improved code MELCOR 1.8.5 is used, and the number of nodes has been increased from 22 to 236. In view of the time needed, MELCOR is used for calculations in the initial phase, then the calculations are continued using specialized codes for containment analysis. SAMG analytical validation followed choice of scenarios selected by means of probabilistic analyses (PSA1, PSA2). Validation matrix of scenarios included

- Basic scenarios
- Scenarios with accident management measures according to SAMG

Criteria for evaluation of measures’ effectiveness included consideration of

- Reaching of long term stable state
- Reducing of activity releasing

- Reduction of barrier breach risk
- Reduction of high energy phenomena occurrence (in containment)
- Timing of major event (to allow AM measures implementation)

Assessment of AM measures was based on results of performed analyses of hydrogen distribution in containment.

Comprehensive project in progress (2004–2007)

It includes development of methodology for analytical assessment of hydrogen flammability using a detailed containment nodalization with the following characteristics:

- 79 nodes and 193 flow paths in lower containment part
- 157 nodes and 456 flow paths in reactor hall.

The nodalisation used in the improved containment model is shown in the drawing below.

A multi-compartment containment model was used for MELCOR 1.8.5.

Calculation of 2 representative scenarios is to be finished by the end of 2006. The calculations will include criteria for flame acceleration and transition from deflagration to detonation (compartment shape, H_2 concentration, mixing …). In the result, assessment of possible hydrogen risk will be made and additional corrective measures to prevent containment failure due to hydrogen deflagration or detonation will be determined (if needed). According to unofficial expectations, the number and capacity of hydrogen recombiners will be increased. The decisions are expected in the end of 2007. Until then the containment of Temelín will be provided with the existing recombiners, which are sufficient for the purpose of control of hydrogen arising in the case of design basis accidents.

SAMG Validation

The following basic strategies used in SAMGs in Temelín are of particular importance (the guidelines on which they are based are given in each case):

Inject into steam generator (SAG 1)

Goal of measures:

- Prevent SG tube creep rupture
- Reduce fission product releases by water layer

Scenarios considered:

Basic scenario: ZC-1 – SGTR, equivalent diameter 40 mm (multiple tube rupture in upper row)

Scenarios with AM measures

- VC-1/2 – inject into SG 5 minutes after core exit temperature exceeds 650 °C, level increase to 225 cm (just covering tubes)
- VC-1/5 – inject into SG 5 minutes after core exit temperature exceeds 650 °C, level increase to 320 cm (100 cm above tubes)

A comparison of releases from SG and decontamination factors (DF) (presentation “Temelín Accident Management Programme”, see Appendix 1 for full reference) is shown in the table below.

Mass [kg] Class	ZC-1 Release	ZC-1 DF	VC-1/2 Release	VC-1/2 DF	VC-1/5 Release	VC-1/5 DF
XE	272.200	1.00184	285.300	1.00030	289.800	1.00003
CS	23.020	1.43393	7.952	4.89713	6.773	6.39790
BA	0.205	1.41857	0.180	2.91991	0.185	4.79320
I	0.387	1.00417	0.505	1.64707	0.667	1.79170
TE	8.831	1.44446	6.869	2.24778	5.221	3.10496
RU	0.540	1.38658	0.140	5.50358	0.133	6.63568
MO	22.880	1.40559	7.330	5.29877	7.060	6.53399
CE	2.61E-05	1.41843	1.04E-05	6.26174	2.13E-05	6.91165
LA	2.38E-05	1.41763	9.51E-06	6.23975	1.93E-05	6.91003
U	27.980	1.38670	6.979	5.42614	6.423	6.60330
CD	0.122	1.42399	0.035	5.13074	0.032	6.49056
SN	0.474	1.40645	0.145	5.30138	0.140	6.62616
CsI	13.040	1.22393	13.830	1.23514	11.840	1.41334

Containment pressure reduction (SAG 6, SCG 2)

Goal of measures:

- Containment pressure decrease with loss of containment spray system
- Reduce fission product releases from containment

Scenarios considered:

Basic scenario: LOCA 200 mm with loss of core cooling

Scenarios with AM measures

- Pressure reduction using HVAC TL01
- Pressure reduction using HVAC TL21
- Pressure reduction using fire protection system UJ

A comparison of source terms with and without containment failure (presentation “Temelín Accident Management Programme”, see Appendix 1 for full reference) is shown in the table below.

Class	Fraction of Initial inventory [%]				
	Non-Failed	Failed			Difference
		Oper Leak	Break	Total	
XE	0.1779	0.1776	2.1970	2.3746	2.1967
CS	0.0056	0.0055	0.0006	0.0062	0.0006
BA	0.0002	0.0002	0.0009	0.0011	0.0009
I	0.0621	0.0621	1.3230	1.3851	1.3230
TE	0.0088	0.0088	0.1250	0.1338	0.1250
RU	0.0005	0.0005	0.0002	0.0007	0.0002
MO	0.0056	0.0056	0.0009	0.0065	0.0009
CE	0.0001	0.0001	0.0002	0.0003	0.0002
LA	0.0000	0.0000	0.0000	0.0000	0.0000
U	0.0001	0.0001	0.0000	0.0001	0.0000
CD	0.0064	0.0064	0.0011	0.0075	0.0011
SN	0.0058	0.0058	0.0011	0.0069	0.0011
CsI	0.0103	0.0103	0.0105	0.0208	0.0105

Containment failure at pressure of 800 kPa (ekv.diam. 0,1 m^2)

(Pressure with 5 % probability of containment failure)

Staff training

Staff training is conducted regularly, with classroom session and simulator work.

Class room session

- AM and emergency preparedness
- SAMG structure and strategies
- Plant capabilities
- SA phenomena and analyses results

Full scope simulator training is provided both for operating staff and TSC personnel. It covers

- Training of extraordinary events (Alert, Site Emergency and General emergency level)
- Usage of EOP and SAMG
- Cooperation of simulator, TSC, Emergency commission, Emergency Operation Center, Emergency support center …
- Special training in SAMG usage coordinated by Westinghouse experts (in preparation)

Plant improvements

Features of existing Temelín design

- Requirements to instrumentation (operability, availability of information)
- Combustible gases management – passive autocatalytic recombiners (PARs) in containment. As of now, in Temelín there are 22 Siemens type PARs (FR 90-1/150). They are shown in Fig. 4a, 4b in Appendix 3. Four of them are situated in the containment at middle height of the containment dome, the remaining are distributed in the containment compartments below the reactor hall floor (presentation “Temelín Accident Management Programme”, see Appendix 1 for full reference).
- Reactor coolant system depressurization capability (pressurizer PORV (pilot operated relief valves), YR venting)

Additional design changes

Decrease of MCCI progress: It is required that the basemat should not fail over 24 hours after the accident. However, there is no analysis available to demonstrate that the doses will be small, if the failure occurs after a time longer than 24 hours after the accident. In answer to a direct question, Czech side acknowledged that the codes available are not good enough to predict such situation, and the experimental investigation of basemat penetration conducted in EU states has not been finished yet. The technical means used to prevent early basemat failure include:

- Stuffing of ex-core ionization chambers channels (this measure has been designed and decided; it will be physically implemented during the next outage, 2007)
- Enlargement of area for molten corium after reactor vessel failure

Measures to permit molten corium spreading also concern the doors between reactor cavity and the containment. There is a shielding door on the inside, which will be permanently opened during reactor operation; and a hermetic door on the outside, which is closed during operation and either will be opened timely during an accident sequence, or will be provided with an insert of easily melting material to allow the molten core material to spread into the containment. It is still under consideration which measure will finally be taken.

To the question as to what would happen if the hermetic door were closed (with an insert) at the time of RPV failure, the Czech side replied that that door might be damaged by the pressure, but it was considered as likely that it would not fail altogether. Hence, there could be no ejection of core debris into the containment.

(In this context, the following should be noted: If, in case the door is kept closed, the insert does not guarantee sufficient spread of melt, basemat penetration will be much faster than in the case the door is opened.)

In order to protect the containment wall some facilities situated inside the containment against possible overheating and melting under the influence of molten corium ejection, removable barriers against corium will be installed.

Enlargement of coolant inventory inside containment will support effective corium cooling. 150 m^3 of additional coolant will be stored in the cavity for reactor barrel inspection (which is currently empty during normal operation). There is already a line, with a valve, for cavity drain. A new valve will be installed in parallel for redundancy.

Hydrogen counter-measures: Parameters of PAR presently installed:

- Temperature range 29 – 131°C
- Pressure range 10 – 43.2 MPa
- H_2 concentration 0 – 14.1 % obj.
- Steam conc. 0 – 68 % obj.
- Seismic qualification
- Capability 52 g H_2/h
- Recombination initiation 2 % vol. of H_2 (100 kPa, 50 °C)
- Recombination termination 0,5 % vol. of H_2

Filtered venting system operation has been checked, and the resistance of filters to heat due to accumulation of fission products is confirmed. (This system proved a high pressure containment venting option through the normal operation stack filters.)

Revised procedural framework is implemented as planned (E-plan, Emergency Implementation Procedures and EOP)

Intentional releases of radioactive effluents in emergency conditions (as one of the SAM strategies) are clearly defined and understood within the overall emergency response organisation. The releases would be proposed by technical support team and accepted (or not) by the decision makers in the technical support center (TSC). No other authorizations are deemed necessary.

SAMG validation process is completed including observations on sufficiency of staffing and organisational structure of TSC (SAMG Evaluation Group) and related feedback.

Staff training in the area of SAM is regularly held 3 times a year.

The capacity of batteries is 2 hour, it has been checked by GRS and found sufficient according to contemporary safety practice.

Emergency control room

The emergency control room (ECR) supports reactor shutdown and cooldown in the cases when the Main control room (MCR) is disabled. Shift staff abandons MCR and enter ECR when one of the following hazards occurs:

- External hazard (e.g. physical risk to MCR and other external effects disabling plant operation from MCR)
- MCR inhabitability (loss of HVAC systems; radiation hazard; fire, smoke, steam or toxic gases, etc)
- Loss of control from the MCR (failures of I&C, control or parameters display)

Design principles for ECR in Temelín are based on the standard of the International Electrotechnical Commission „Supplementary control points for reactor shutdown without access to the main control room" (IEC 60965 Ed. 1.0 b, 1989). It fulfils the following requirements:

- ECR provides means for reactor trip and verification transfer to cold state including maintaining it
- ECR provides sufficient information for assessment & surveillance over shutdown and long-term core cooling
- Cold shutdown shall be achieved within 72 hours also with a common cause failure in the I&C system
- Single failure criteria apply to ECR

Requirements for the ECR:

- ECR shall be habitable for a time necessary to perform the unit control from ECR
- ECR assures environmental conditions and protection against radioactive exposure similar as MCR
- No operator intervention needed for transfer from MCR to ECR
- Emergency procedure shall define the transfer from MCR to ECR and further activities in the ECR
- All equipment that is a portion of safety systems shall be seismically qualified
- Redundant safety equipment shall be physically separated by the fire barriers

In Temelín ECR provides workplaces for four operators, namely unit supervisor, control room supervisor, primary side operator, and secondary side operator.

Unlike in MCR, individual workplaces are not separated.

The ECR is fully separated from MCR (space, ventilation, power supply), it has independent control systems which can execute all functions needed for safe reactor shutdown.

Containment isolation – leakages through small pipes

Containment safety functions:

Envelops all primary system elements and protects them from external hazards

Protects NPP environment against radiological hazards during normal operation and during accidents

Prevents releases of radioactive products during accidents and helps to reduce concentration. This is achieved by natural processes (like radioactive decay and deposition) and action of special safety systems.

Isolation of all penetrations is a key feature for operability during accidents.

Temelín containment is a classical, large volume, full pressure single wall structure made of pre-stressed concrete with an internal steel liner, with **very low leakage rate** (tested). **All penetrations** are pressure rated and **piping penetrations** are equipped with series of automatically-operated isolation valves. However, **safety system and some small-bore instrument** lines are not automatically isolated.

Containment wall is penetrated by numerous pipes of various diameters, from dozen of cm (i.e. steam line 800 mm nominal diameter) to a few mm (i.e. impulse line 12 mm). **Most of those are equipped** with fast acting automatic valves, which will isolate pipes from the external environment. **Pipes of safety systems** situated outside the containment are not automatically isolated, because safety systems cannot be cut off from the reactor.

In comparison to the original Russian design, Temelín NPP has been changed in many respects. One of the new features is Post Accident Monitoring system (PAMS), provided according to the requirements of US NRC. It includes 5 sampling lines, 3 of them taking samples of the containment atmosphere and 2 taking samples of the liquids. According to the original Russian design, all these lines should be automatically closed in case of accident, because no PAMS was required. Therefore their parts located outside containment are not qualified to accident condition, In case of sudden increase of pressure or temperature they could fail. However, according to the new design of Temelín, PAMS lines should be open so that sampling can be done. This means that automatic closing is excluded and the lines should be closed by the operator and then opened as needed when the pressure and temperature are low enough. However, this involves hazard of leakages outside the containment.

If valves on PAMS lines remain opened, fluids containing radioactivity would reach sampling equipment. **This equipment is not qualified** for accident conditions.

If equipment boundaries are breached, radioactivity will be released into controlled zone (within which automatic system monitors the radiation level). **Upon observing increased radiation level**, operator is expected to close all PAMS valves. **Even if PAMS valves are left opened**, there will be no direct release. Controlled area is filtered, to retain radioactive aerosols.

Recent resolution by SÚJB

As agreed in July 2006, 9 valves on those three PAMS lines which are connected with the containment atmosphere should be permanently closed.

The remaining 6 valves on PAMS lines connected to the liquid media inside the containment are related to systems which are needed in normal plant operation and therefore cannot be permanently closed.

In an accident, those will be immediately closed by the operator, in accordance with the emergency operating procedures.

Evaluation

Generally the information provided by the Temelín NPP shows that the work on safety upgrading of the plant is being continued, and in some areas the progress is significant. In the area of Severe Accident Management the analyses are continued with improved calculation tools, and are expected to be completed in 2007, after which some upgrading of hydrogen recombiners system is to be expected.

The issues connected with organization of Severe Accident management have been generally solved and proper training has been assured. There is progress in the development of the design of technical measures needed for severe accident mitigation, and the implementation is under way (presentation "Temelín Accident Management Programme", see Appendix 1 for full reference). However, the process is not finished yet and deserves further monitoring.

Issues of Further Interest:

The completion of severe accident analyses, expected in 2007, should be further monitored.

Also, the further development regarding the design of technical measures deserves monitoring. In particular, this concerns the following points:

- Upgrading of hydrogen recombiners
- Measures for enlargement of the molten core area:
  - Reactor cavity doors – inside (shielding) door permanently opened during operation; outside (hermetic) door opened timely during accident sequence, or provided with insert which melts easily
  - Removable barriers to protect containment wall from corium
- Stuffing of ex-core ionization chambers' channels (to be implemented 2007)

- Enlargement of coolant inventory inside containment (it is not clear whether this measure will have a significant positive effect since analyses show that even if the reactor cavity is filled with water and the external area covered with water, basemat penetration will occur practically unaffected)

When the complete solution is finally being implemented, it would be of interest to review it from the Austrian side. The final (implemented) solution would need to be verified against calculations that will provide the results on the velocity of melt-through (MCCI) with the enlarged area. This is being impacted by both the solution to be implemented for the outer door opening (mechanical or passive), but also the positioning of the barrier walls.

5 SEISMIC ISSUES (ITEMS 6, "SEISMIC DESIGN"; PN 6, 8)

(Contributed by VCE)

Introduction

The Final Monitoring Reports (May 2005) as well as the Summary Monitoring Report (June 2005) specify several topics related to Items 6 and "Seismic Design" where a follow up is useful and necessary. The topic seismic monitoring system has been chosen by the Czech side for this meeting. The Austrian side presented new information on the evaluation of site effects and vulnerability assessment of buildings and structures.

Some comments to this Item were also provided by IRR. They can be found in Appendix 2 to this report.

Summary of New Information Provided

The presentation by the University of Brno on the micro-earthquake monitoring-upgrade of the seismic monitoring system has been received. It provides details on the new instrumentation and the layout. Furthermore, the first results were presented. The Austrian delegation was able to visit the installed system outside the plant.

Only a sample of monitoring results could be provided because the system is not in operation long enough. No detailed information could be presented yet. The new system replaces an old system which operated from September 1991 till December 2005. The new system is based on the IAEA 2003 expert mission recommendation and has been financed from the state budget of the Czech Republic. The operation started in January 2006. The system can be operated by remote control. During the short operation time no major earthquake event was recorded. A record of a typical quarry blast has been presented.

An Austrian presentation on seismic site effects and structural assessment has been provided with the related quotation of the relevant IAEA standards. Major steps in development enable now a considerably improved seismic vulnerability assessment of structures, based on the results of successful European research projects (RISK-EU and SESAME). The conclusion is that by relatively simple field measurements valuable data for a proper Probabilistic Seismic Hazard Assessment (PSHA) can be obtained. This has relevance to the coming 10 year periodic inspection report, where the seismic subject shall be reopened according to IAEA regulations and the recommendation of the FMR.

The Austrian presentation is added to this report in full length as Appendix 4.

Evaluation

The new monitoring system constitutes a clear improvement. This topic has been upgraded satisfactorily. Still open is the evaluation of the obtained data in order to come to a more accurate – measurement based – assessment of the seismic hazard for the Temelín NPP site.

Issues of Further Interest:

Only one specific topic has been touched in detail in this meeting leaving all the other topics of the FMR and SMR (2.6 and 2.7) open for further bilateral contacts and clarification. The recommendations brought forward in the Austrian workshop presentation are in line with the recommendations of the SMR and were meant to improve the understanding of these issues. When the seismic issue will be opened in the course of the 10 year safety review it shall be checked whether the new recommendations have been adopted and implemented in the promised probabilistic seismic hazard assessment to be performed.

It has been further expressed that it would be of interest to receive raw data of the monitoring system in order to carry out evaluations beyond the scope of the normal operation of the system.

Preparatory Work for Future Discussions:

As mentioned in the previous chapter it would be most useful to perform an assessment of the monitoring data of the new system. This assessment could bring valuable information on regional seismic properties such as seismic impedance and wave speeds.

The Czech colleagues from the University of Brno have showed interest in this collaboration and promised the submission of data in case that the plant management agrees. At the meeting there has been no objection from the plant operator.

6 CONTROL RODS AND FUEL DEGRADATION (NEW ITEM)

(Contributed by IRR; Addendum by ENCO, IRR and Physikerbüro Bremen)

Introduction

In the context of the Extra budgetary Program for the Safety of WWER-1000 NPPs the IAEA has defined the Issue RC2, "Control Rod Insertion Reliability/Fuel Assembly Deformation", and ranked it as a Category III issue. (Category IV being the highest rank of safety concern)

Issues in Category III are of high safety concern. Defence in depth is insufficient. Immediate corrective action is necessary. Interim measures might also be necessary.

(For more details concerning the history of this issue, regarding treatment by IAEA as well as between Czech and Austrian experts, see the addendum to this section.)

In Temelín NPP, increasing numbers of rod control cluster assemblies (RCCAs) were unable to touch down during 2004–2006. This topic was dealt with in a presentation on “Actual Fuel Problems” at the workshop.

In addition to this section, further technical remarks, questions and answers received can be found in Appendix 2 to this report (detailed report by IRR).

Summary of New Information Provided at the Workshop

In the SÚJB report of 14 June 2006 “Degradation of nuclear fuel at Temelín Unit 1”, a number of topics of interest are discussed.

Increasing numbers of rod control cluster assemblies (RCCAs) were unable to touch down during 2004–2006; these numbers were communicated in the SÚJB report.

The operator has stated in its presentation, that all precautionary measures have been taken to gain control over the RCCA insertion problem and eliminate eventually the root causes related to the malfunction.

The information provided on the rod insertion time elapsed and the “bottom” distance was qualitative: With the exception of two RCCAs exceeding the 3.5 s time limit, all RCCAs have in all insertion cases remained below this limit, and all RCCAs in all positions and all cases of have passed the OLC position with the absorber zones’ lower end. (The “drop time” is the time that elapses during a cram for an individual RCCA while travelling between the withdrawn positions to passing the OLC position.)

Immediate action was taken when the requirement of the insertion time to remain below 3.5 s, as it is included into the limits and conditions, was not met by 2 RCCAs: The reactor was put out of operation.

The requirement that all RCCAs ought to reach the fully inserted position in order to declare the reactor's state "in safe shutdown condition", could be fulfilled with the motor operated spindles of the control rod cluster drive mechanisms driving the "stuck" RCCAs to the final position. This is not in contradiction with the licensing requirements. Corrective measures were implemented, taking into account these circumstances by imposing strict requirements for testing and analyses for the period these concerns persist.

The presentation by a representative of the operator, "Temelín NPP – Actual Fuel Problems" (see Appendix 1 for full reference), provided information on:

- Problems with RCCAs (Control Cluster Rod Assemblies) during operation
- Fuel Assemblies (FA)/Fuel Rod (FR) Bow

It was stated that mechanical deformations of FA and FR come as a natural phenomenon during reactor operation. Fuel assemblies bow and twist because of radiation growth of FRs and the FA skeleton.

Changes in the core fuel reload pattern performed since the last workshop information were presented. FA bow affected refuelling of the core. The changes consist in replacement of one fuel assembly, and replacement of the RCCAs containing FAs. Later, a change to a new version of fuel (different fuel supplier with more experience in WWER-1000/320 fuel) is envisaged. (In the meantime, it has been publicly announced that a new contract to supply fuel for Temelín starting in 2010 has already been signed by ČEZ and the Russian fuel supplier TVEL.)

However, FA bow and twist are not seen as a safety problem by the Czech side; rather, they are regarded as usual phenomena in all reactors in the world.

The issue of leaking FAs was also addressed. Since fuel leaks are not important for the RCCAs' movements in all cases, however, they are dealt with in section 7 below.

The presentation also covered the details concerning the CCRA insertion problems analyses, the assessment for the safe operation of the plants under these conditions, consequences and remedies identified and the plans when and how the related measures will be introduced, tested and approved.

FR bow was reported to be caused by different compressive forces due to spacer grids used to align the FRs. The magnitude of mechanical deformations is limited by the cross sectional area left available for the coolant to flow through. The first cores in Temelín were designed to tolerate flow area reductions by 51 %; all subsequent core designs are calculated with the conservative assumption that the FRs could touch each other.

Evaluation of Workshop Information

There is no physical limitation to the mechanical deformation by the flow area. The physical limitations are set by the touching of the fuel rods, may there be one bent or two or a whole group of them "leaning" against each other.

The geometry of the problem is not only 2-dimensional. A 3-dimensional limitation is required, to be verified by ISI, in that the vertical extent of the FRs touching area is of interest, as well as the "geometrical rearrangements" of entire groups of FRs. These "rearrangements" are limited to the volume between two adjacent grid spacers within each FA, and they have consequences to the normal operational heat transfer and to the emergency core cooling situations as well.

The very simplified description of improvements requires a more detailed supplement describing the factual limiting conditions and how these can be verified for the requalification of the FAs for further service.

However, the progress reported in such areas as immediate actions, corrective actions and preparations for root cause elimination is in line with the actions taken at other plants (Kozloduy NPP, Balakovo NPP etc.). Even though these measures were successful, only after their combination was optimized, it can be expected in this case, too, that a solution can be found.

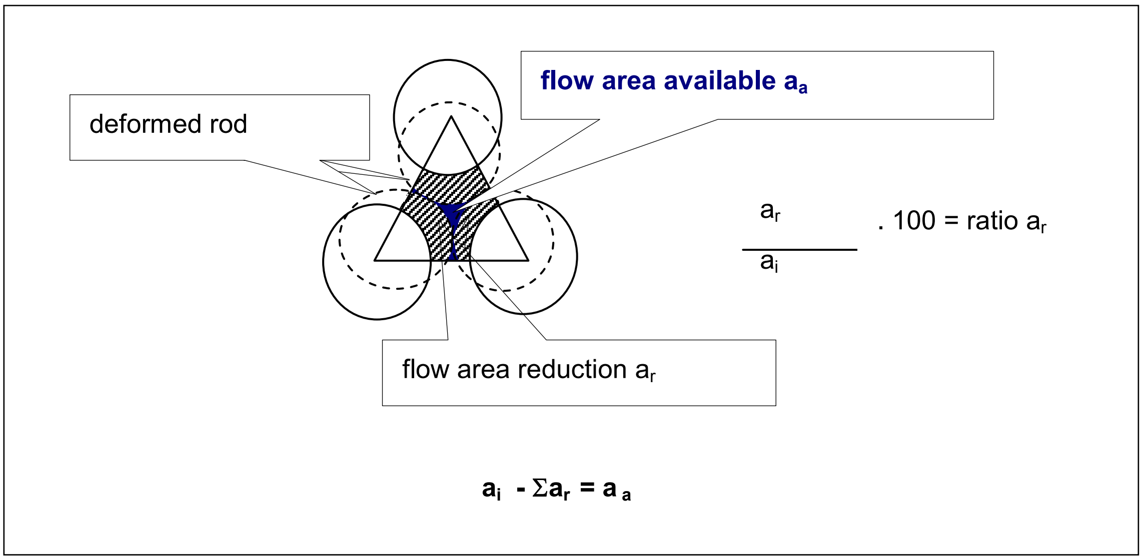
Changes are envisaged in the core and regarding the fuel assemblies in general as well as those fuel assemblies where the RCCAs are inserted. These changes will eventually contribute to a definitive solution. In this respect, however, the expectations' fulfillment still needs to be confirmed, once the new equipment is in place and has been operated at least for some time.

Before that the Austrian team of experts expects precautionary administrative and testing measures to remain in place for the time period when shortcomings identified before still persist.

Issues of Further Interest as Identified from Workshop Information

There are open questions regarding the details of the factual limiting conditions of FR deformation, and how those conditions can be verified for the requalification of the fuel elements for further service:

- Does the permissible state of the FRs signify an arrangement as shown below in its cross-sectional representation? Or is the touching zone formed by just two adjacent FRs?



- How would, in both cases, this touching zone extend between two adjacent spacers?
- How does this representation relate to the statement of the Czech side *"… the conservative assumption that the FRs could touch each other"*?
- What are the limits imposed on the axial extent of the touching zone?

Addendum – Questions to be Followed Up

After the Workshop in Temelín, additional considerations were performed by Austrian experts regarding the questions to be followed up in connection to control rod insertion problems.

The results of the additional work are summed up in this addendum.

Introduction

Control rod insertion reliability has been identified as a safety issue for WWER-1000/320 nuclear power plants in the early 1990s.

The International Atomic Energy Agency (IAEA) has discussed this issue in the Report IAEA-EBP-WWER-05 (Safety Issues and Their Ranking for WWER-1000 Model 320 Nuclear Power Plants, March 1996), generally known as "Issue Book". The issue was ranked in Category III (high safety concern; immediate corrective action necessary, interim measures might be necessary). A number of compensatory and interim measures are listed in this report (see below). It was emphasized that the experience to verify the design modifications by normal operation was not sufficient at that time, and the root cause was not fully established.

The issue was referred to again in the Report IAEA-EBP-WWER-15 (Final Report of the Programme on the Safety of WWER and RBMK Nuclear Power Plants, May 1999), with the gist that its significance had been reduced:

"Currently, the safety significance of this issue is considered to be low since remedial actions have been taken, long term corrective measures are underway, defence in depth provisions exist in the reactor design, and adequate safety margins remain. However, the industry should continue data collection on the problem and continue with the implementation of corrective measures."

The situation at Temelín NPP was explicitly mentioned in this report. It was clearly referred to as being under control because of the change of the fuel supplier (from a Russian producer to the U.S. firm Westinghouse):

"The issue is considered to be properly addressed by hardware measures taken and planned follow-up actions adopting WEC technology at Temelín NPP."

The issue was further discussed in the course of expert consultations between the Czech Republic and Austria, with the participation of the European Commission ("Trialogue") in 2001. The Czech side provided the following explanation:

"The problems experienced with the Russian design should not occur with the new Westinghouse design."

It was also emphasized that monitoring and verifications were to be carried out to further observe and assess the situation.

The issue of control rod insertion reliability was not regarded as closed; it was found suitable, however, to be followed up in the framework of the pertinent bilateral Czech-Austrian agreement. Therefore, it was not part of the safety Items of special importance which were listed during the Melk Process (Annex I to the “Conclusions of the Melk Process and Follow-Up”, November 29, 2001) and which were followed up in the monitoring on a technical level until 2005.

Recent developments show, however, that the issue does not appear to be completely resolved in Temelín. Problems with RCCAs (Rod Control Cluster Assemblies) have occurred at both units of Temelín NPP during operation.

Repeatedly, RCCAs did not achieve full touch-down in the bottom position, although they kept the required drop time (3.5 sec) to the level of the hydraulic dampers, as prescribed by the Limit Condition. Furthermore, at the occasion of a test on June 02, 2006, two RCCAs at Unit 1 stopped above the level of the hydraulic dampers and thus failed to meet the Limit Condition (i. e. a requirement which has to be fulfilled at all times in the interest of safe operation). As a consequence, operation of this Unit was immediately stopped by the supervising authority SÚJB (State Office of Nuclear Safety).

The number of RCCAs failing to fully touch down, in Unit 1, showed an increasing trend during the course of the last two fuel campaigns (the problem is less critical, so far, at Unit 2). At the test on June 02, 2006, 51 RCCAs (out of a total of 61) did not achieve full touch-down (including the two RCCAs failing to meet the Limit Condition).

The following tables provide an overview of the numbers of clusters failing to touch down in tests conducted in 2005 and the first half of 2006 (provided by SÚJB, www.sujb.cz):

Unit 1:

	3rd campaign					4th campaign						
Date of testing	1.1.	27.3.	30.3.	14.6.	30.7.	4.10.	19.11.	30.12.	25.2. 06	17. 3.	7. 5.	2. 6.
Number of clusters	11	12	12	21	30	2	13	18	32	33	45	51

Unit 2:

	2nd campaign					3rd campaign	
Date of testing	5.2.	12..3.	9.4.	15.7.	3.9.	6.1. 06	1.5. 06
Number of clusters	14	14	17	0	0	0	0

Thus, the insertion performance of a remarkable number of RCCAs is impaired, with some cases where the prescribed Limit Condition was not kept. This constitutes a case of incomplete rod insertion at Temelín NPP. It appears to be caused by fuel assembly (FA)/fuel rod (FR) bow.

Information on these problems, as well as on counter-measures implemented and planned, had already been provided by the Czech side at the occasion of the bilateral meeting in Vienna, November 15, 2005. Further information was provided during the walkdown/workshop in Temelín at September 26/27, and the bilateral meeting in Prague at November 06/07, 2006. This information was instructive and helpful to the Austrian side and has significantly contributed to the understanding of the current problems, and to the assessment of their relevance.

However, open questions remain. The Austrian side would appreciate further explanations and clarifications because the system involved – the fast shutdown system of the reactors – is of crucial importance to safety and hence, significance has to be attached to all problems concerning this system (hence the ranking in Category III by the IAEA's EBP-advisors).

Furthermore, it is known that Westinghouse fuel assemblies of different design (square cross-sections) have also been observed to be particularly vulnerable to deformation. It will take at least four years until the bowed fuel elements in Temelín can be replaced completely by fuel elements from a Russian manufacturer. Thus, problems with Westinghouse fuel are likely to persist in Temelín for some years to come.

In view of the recent resurgence of the control rod insertion issue at Temelín NPP, in spite of the IAEA's judgment in 1999 and the assessment provided in the course of the "Trialogue", it seems appropriate to take up once again the measures formulated in the "Issue Book" in 1996, investigate the extent of their application at Temelín, and to also consider questions which go beyond the 1996 IAEA requirements.

Questions Concerning the Control Rod Insertion Problems:

The five requirements listed in the IAEA's "Issue Book" of 1996 provide a starting point. They are listed here, together with comments and questions which relate to the situation at the Temelín NPP.

1. *If excessive rod drop times are observed at full coolant flow rate, operation with three or two reactor coolant pumps at correspondingly reduced power is permitted, provided that the measured drop times of any rod does not exceed 4 seconds. If the transfer to operation with three or two coolant pumps is not successful, then the unit has to be shut down.*

It seems clear that this requirement was not applicable in Temelín after the test on June 02, 2006, when the Limit Condition was not met. In this situation, the appropriate reaction was to stop the operation of the unit – as the supervising authority SÚJB did.

In principle, the requirement could have been applied before June 02, 2006, if "failure to touch down" would have been taken as equivalent to "excessive rod drop times". It would be of interest to know whether this was considered by the operator and/or the authority, and if so, why reducing coolant flow rate was not implemented.

2. *Control rod drop times are measured at least once every 3 months. If any control rod drop time is more than 4 sec, the next test is carried out within a month.*

In Temelín, control rod drop time tests were generally performed, according to operation specifications, in intervals of a few months. Tests were conducted, additionally, every time a transient had led to reactor shut down. Starting in 2005, the test intervals were reduced, finally to an interval of 30 days, for cycle 5 of Unit 1. This interval has been specified by the authority after the test at June 2, 2006. The detailed considerations leading to reduction of the test intervals would be of interest.

Furthermore, the core coordinates of the RCCAs not touching down would be of interest, as well as drop time to the position corresponding to the Limit Condition (as far as applicable).

The drop time limit (for the rod to reach the positions specified by the Limit Condition) at Temelín NPP is 3.5 sec. It would also be of interest to know which differences in core and/or fuel design is the reason of this reduction, compared to the value of 4 sec mentioned in the first requirement.

3. *In order to minimize the potential rod insertion problems, fuel assemblies which have been used for 2 years are not inserted into the control rod locations, but are replaced by new fuel assemblies with nearly the same physical characteristics.*

It is known that the insertion problems tend to increase with increasing burn-up. It would be of interest to receive information concerning the burn-up of the specific fuel assemblies where the problems with control rod insertion at Temelín occurred, to verify whether there is indeed a connection to burn-up. Furthermore, the operating history of the fuel elements used with RCCAs would be of interest.

The question arises why the insertion problems have increased with cycle numbers. Has the IAEA recommendation to change the FAs with RCCAs after two cycles been followed at Temelín? If not, what were the reasons not to introduce this policy? Were there any changes in the refuelling pattern during 2006?

4. *Before loading of fuel assemblies into the core, they are tested on stands for verification of free control rod movement. The deviations of lifting and lowering forces from normal values should not exceed ±3 kg. The central instrument thimbles are measured by means of a specially designed calibre.*

Which tests identifying deviations of lifting and lowering forces have been performed, what were the results? Which deviation limits have been found applicable for Temelín NPP (since the value given here might not be appropriate for Westinghouse fuel elements)?

5. *The position of the upper internal structure (protective tube unit) was readjusted and moved upward for several millimetres to reduce the excess axial load exerted on the fuel assemblies and to alleviate the deformation of guide tubes.*

This measure has, for example, been applied at Kozloduy units 5 and 6.

A reduction of the spring forces acting on the pellets in the fuel rods from both ends constitutes another possible measure. This has been briefly discussed at the workshop in Temelín at September 26/27, 2006. It was pointed out by the operator that it is a matter of optimisation – in case of strong spring forces, there is more deformation, but less leakage from the fuel rods; and vice versa in case of weak spring forces.

Furthermore, reducing spring forces could be ineffective against tension building up towards the middle of the rod. Generally, as long as the FA structure is not deformed, processes occurring in the rods are of no consequence. The positioning and numbers of the spacers is of importance regarding deformation (and local flow path reduction).

Further information regarding the technical background of these corrective measures (upward moving of upper internal structure, as well as others like readjusting spring forces) at Temelín would be of interest to the Austrian side.

There are further questions which the Austrian experts consider as relevant in the context of the control rod insertion problems:

1. *Which investigations have been required by the authority to demonstrate that safe shutdown is guaranteed under accident conditions (LOCA, earthquake etc.)?*

In case of LOCA, strong hydraulic forces can act on the fuel assemblies. Damage could occur in case there has been deformation and/or weakening of the FAs before. Similar considerations apply to seismic events.

It would be of interest to the Austrian side which investigations have been required by the authority to clarify whether the observed deformations of the fuel elements

- jeopardize the capability for safe mechanical shutdown under accident conditions,
- indicate some weakness of the mechanical design, endangering this capability, and
- are adequately covered by the testing interval of 30 days, when accident conditions are taken into account.

2. *Which strategy is envisaged for the replacement of Westinghouse fuel with fuel from a Russian supplier? Is it possible to perform the exchange gradually, operating with a “mixed” core for some years? If so, would it contribute significantly to the solution of the RCCA insertion problem if, as a first step, only FAs with RCCAs were to be exchanged? If this applies – would it be possible to acquire a smaller number of Russian FAs from the supplier TVEL before the year 2010 (according to a media report, delivery of Russian fuel is to begin in April of this year)?*

A possible measure which would render the question for the root cause(s) irrelevant would be to simply reduce the time fuel assemblies remain in the reactor, for as long as Westinghouse fuel is still used. There would have to be a sufficient number of Westinghouse FAs available for this option, however; with the side effect that fuel costs would be increased.

There are indications in media reports that this is indeed the strategy to be followed.

It would be of interest to the Austrian side whether neighbouring fuel assemblies which are deformed can influence the FAs containing RCCAs. (If this were the case, it would not contribute much to the solution of the problem if the FAs with RCCAs only are replaced.) Also, the strategy followed so far would be of interest – when refuelling in 2005 and 2006, were all fuel assemblies with RCCAs replaced, or selected ones?

3. *What is the state of knowledge concerning the root cause(s) of FA/FR bow in Westinghouse fuel? Taking this state of knowledge into account – can it be excluded that the situation will deteriorate further, in spite of the first counter-measures which have been implemented? In particular, can it be excluded that RCCAs get stuck in a position higher up in the core, with a greater effect on reactivity? Regarding the change of fuel supplier – what is the state of knowledge concerning the root causes for Russian-made fuel? Can it be excluded that insertion problem occur with such fuel?*

According the international present state of knowledge, incomplete rod insertion is influenced by several factors (fuel element mechanical design, fuel rod design, pellet design, fuel burn-up, coolant temperature, cycle length etc.).

More detailed information on root cause(s) could possibly be made available by Westinghouse. Furthermore, information concerning the root cause analyses for Russian-made WWER fuel would be of interest.

It has to be noted that the counter-measures performed so far (adding weight to RCCA, increasing the size of the holes in the dashpot) do not address the root cause(s).

As long as there is no definite identification, in detail, of the mechanisms leading to the insertion problems, it has to be feared that the situation will deteriorate further. Indeed, the number of fuel elements affected is increasing, according to the data made available to the Austrian side.

4. *To which extent have the limitations concerning loads during the service of components been approached already by the sequences of control rod tests?*

While the loads exerted on the reactor pressure vessel are not likely to be significant, the number of scrams for clutches, motor drives etc., as well as for the lower core plate is limited for thermal and hydraulic reasons.

7 LEAKS FROM FUEL RODS (NEW ITEM)

(Contributed by IRR)

Introduction

Fuel rods are the prime components of a fuel element. For retention of radioactive material the rods are welded leak tight tubes containing the fissile material pellets. Due to the high radiation density in the core the tube material experiences changes in its microcrystalline structure. The fuel can reach high temperatures; internal pressure is high. These loads in combination can result in cladding failures before the end of the total number of fuel cycles the fuel elements have been designed for.

IAEA defined the safety concern with regard to fuel leakage as follows:

Issues of this kind are a safety concern in so far that one of the barriers function to retain radioactive material has failed and the maximum allowable global leakage limits for releases from the fuel apply.

In Temelín, there is an increasing number of leaking assemblies which are identified and removed from the core during refuelling.

In addition to this section, further technical remarks, questions and answers received can be found in Appendix 2 to this report (detailed report by IRR).

Summary of New Information Provided:

The reactors at Temelín contain 163 fuel assemblies each; each assembly holds 312 fuel rods. All nuclear power plants are designed on the assumption that a certain number of rods develop leaks.

The number of fuel rods which are permitted to become leaky is determined with reliance on safety analyses and detailed in Technical Specifications in the form of Safety Limits and Conditions. At the Temelín NPP, the fuel rods are checked for tightness through monitoring of the specific activity in the primary circuit. The overall specific activity limit is $3{,}7 \times 10^{9}$ Bq/l, the limit for the I-131 specific activity $2{,}6 \times 10^{7}$ Bq/l (these indicators are used in all NPPs).

For inspection of the fuel rods' cladding and identification of leaky spots two independent systems are used: On-line sipping and off-line sipping. Based on the results of tests accomplished, several of the fuel assemblies were removed from the cores of Unit 1 and 2 during refuelling outages.

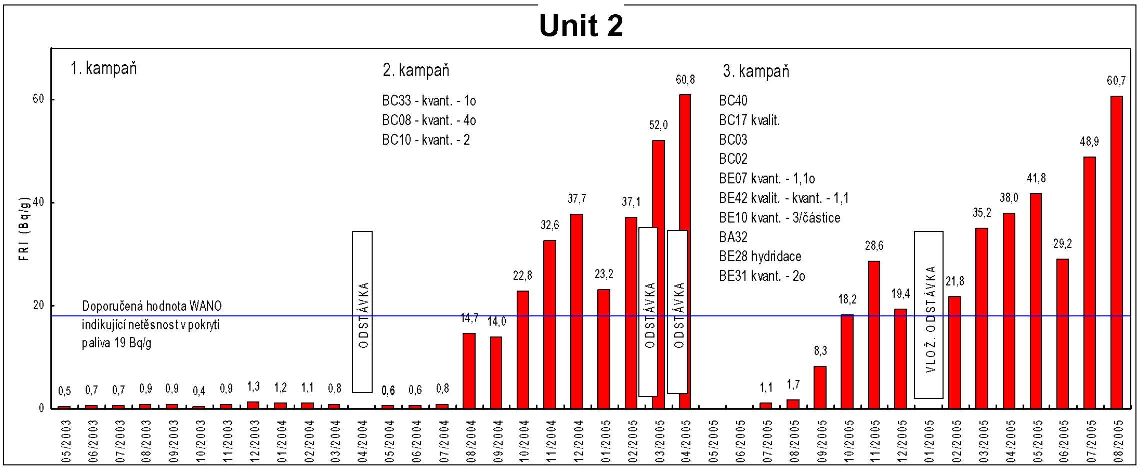


Fig. 7.1: Fuel failure history at Unit 2 developing over 3 campaigns, compared to WANO qualification criterion (presentation, see Appendix 1 for full reference)

It can be seen from Fig. 7.1 that the number of leaky assemblies, identified and removed from the core during refuelling, is increasing with the number of fuel cycles.

The specific activity of the primary coolant during the 4th fuel cycle of Unit 1 (October 2005–June 2006) shows small gas leakages from fuel rods. The overall maximum activity measured in 2006 was 6,4x10^6 Bq/l.

An overview of leaks disclosed in fuel at the Temelín NPP:

Unit / Cycle	Cycle 1	Cycle 2	Cycle 3	Cycle 4
Unit 1	**0**	**1 + 1 suspected FA, but not proven, 2 FAs removed**	**5 FAs, 1 FA repaired 4 FAs removed**	**6 FAs, 1 FA repaired 5 FAs removed**
Unit 2	**0**	**3 FAs / 2FAs, repaired 1 FA removed**	**10 FAs, 2 FAs repaired, 8 FAs removed**	**N/A**

Fig. 7.2: Overview of leaks disclosed in fuel at Temelín NPP (presentation “Actual Fuel Problems”, see Appendix 1 for full reference)

Evaluation

There is no deviation at Temelín from problems as they are commonly encountered with nuclear fuel cladding failures in many NPPs. However, it is worth mentioning that fuel failures with WWER fuel from suppliers originally providing such fuel were not as progressively increasing with the number of service campaigns of the individual FAs.

The extended in-service periods for fuel supplied by Westinghouse could be part of the problem, but also fretting of the grid-spacers caused by the simple clamping type of fixtures used for holding in place, at defined distances, the fuel rods themselves and the control rod guide tubes as well.

Removal of the failed FRs and their replacement by solid stainless steel rods appears to be the practice to deal with the problem. This is an adequate procedure and accepted by the licensing authority.

With respect to safety, there is no concern resulting from the quantities of radioactive material released into the primary circuit. The quantities are well below the permissible limits.

The change of the fuel is expected to also change the failure behavior. Therefore, it is hoped that the situation will be improved.

Issues of Further Interest

Considering the safety relevance to be attributed to the effects of leakage in case of an accident, the number of leaks gives good reason to further observe the development of the leakage rates.

The fuel is supposed to be changed. This should also have an impact, reducing the incidence rate of failures, and should also be observed.

No specific preparatory work is required for the Austrian side to permit further discussion of these issues.

8 RPV HEAD MATERIAL DEGRADATION (NEW ITEM)

(Contributed by IRR)

Introduction

RPV head material degradation is a recognised safety concern with regard to such issues as follow:

Cases of excessive degradation assisted by erosive-corrosive chemical reactions attributed to primary coolant constituents like boric acid as used for reactivity control events, like the extreme one in the Davis-Bessie NPP, cannot be excluded. They would represent the precursor of a large leak in the RPV head, exceeding eventually also the maximum expected leak once the LBB instrumentation does not signal the problem arising well in advance. The event to follow is a BDBA with fast depressurization and extreme flow forces on the core and core barrel.

In addition to this section, further technical remarks, questions and answers received can be found in Appendix 2 to this report (detailed report by IRR).

Summary of New Information Provided

The following information was available on this Item:

- Information on the rusted parts published in the internet (with pictures)
- Information by the operator on cleaning of the corroded parts
- Presentation of the details concerning the boric acid accident and the resulting RPV corrosion
- Presentation of the measures taken: cleaning procedure, NDT evaluation of the RPV components, esp. the RPV head bolts
- Presentation of the measures to prevent similar accidents in the future

At the workshop, the Czech side informed the Austrian delegation that the photos published in the internet had been taken at an occasion much earlier in the plant's history.

During the plant walk-down on September 26th, the Austrian experts were shown the reactor pressure vessel head and the cleaned bolts and nuts. They were informed that all screws had been removed following the boric acid spill event. Due to defects (not explained as to whether they were related or not to corrosion) discovered by NDT one screw was replaced.

Evaluation

The statement of the operator indicates that already several leakage events have occurred that initiated corrosion at the RPV head. This fact demonstrates that LBB (leak before break) might not be fulfilled, due to inadequate leakage detectability. Therefore, no credit can be taken from LBB for the primary circuit integrity monitoring, at least with regard to RPV leaks.

The cleaning procedures performed were obviously effective; according to the Czech experts NDT analyses have been performed.

All reported ISI and maintenance steps are common practice, besides the point that detectability of leakages at the RPV head is evidently not too easy a task for the sensors or even the system installed at ETE. It therefore is to be expected that aside from a check on feasible improvements administrative action will be introduced to help avoiding excessive spills, leakages and consecutive corrosion.

It seems that the operator has recognized improvements needs for the housekeeping.

Issues of Further Interest

The following issues should be further monitored:

- Improvements of leak-detection capability in the primary circuit, particularly the RPV.
- Administrative measures which will be introduced to help avoiding excessive spells and leakages.

No specific preparatory work is required for the Austrian side to permit further discussion of these issues.

9 EMERGENCY DIESEL GENERATORS/RELEVANCE OF FORSMARK-EVENT FOR TEMELÍN NPP (NEW ITEM)

(Contributed by ENCO)

Introduction

Should the power to Temelín NPP be cut-off or a regional electrical grid collapse occur, onsite emergency diesel generators will start automatically to provide power to safety distribution panels. These panels supply power to emergency pumps, valves, fans, and other components that are required to operate to keep the plant in a safe state, or mitigate consequences of an accident.

Further Information on this Item, particularly a detailed description of the Forsmark-event, can be found in Appendix 3 to this report (detailed report of ENCONET).

Summary of New Information Provided

The concern about uninterrupted power supply in Temelín was raised partially in reflection to international interest on the subject matter, as a result of the event which occurred in Forsmark-1 in Sweden July 25, 2006. At Forsmark, through a complex series of events, a short circuit in the switchyard led to the loss of two out of the four trains of safety-related alternating current (AC) and direct current (DC) power due to a common mode failure. The significance of this event lies in the fact that it could have caused a common mode failure of all four trains that could have resulted in the loss of all four trains of safety-related AC and DC power, possibly leading to a severe accident sequence.

The analysis performed by Temelín experts that was presented to Austrian experts showed that although the equipment in question is generally similar, the system differences between Temelín and Forsmark are significant and the scenario of Forsmark could not occur in Temelín.

The redundancy features are different, because Temelín has a 3 x 100% arrangement, while Forsmark has (generally) 4 x 50% redundancy on safety systems. The results of comparisons are presented in a letter of the safety dept of ČEZ, to the SÚJB deputy chairman, send on 25 August 2006 (in Czech). The analysis showed two important differences that would apply to the consequences to a similar initiating event. First of all, just before the accident the system in Forsmark operated in the “fast charging mode”, which involves very high voltage of about 270 V and is not allowed in Temelín, where the voltages are in the region of 220–230 V. Due to this very high voltage, the voltage oscillations were placing the equipment in the region where failures could occurred. Secondly, in Temelín there is not only 3x100% redundancy in the system trains, but in addition there are two inverters-converters sets in each redundant train. One of those is in operation, while the other is in reserve (standby). Even in the case of a failure of one inverters-converters, the other would be available to be switched on and no loss of the system would occur (Letter of ČEZ safety dept. to SÚJB of August 25, 2006; see above).

As the circumstances of Forsmark event have not been fully clarified yet, the analysis of Temelín experts cannot be regarded as final. The work will go on and the results will presented step by step. For the moment however, there are no indications of any hazard of a scenario such as in Forsmark. Moreover, the attachment to the letter shows several curves of voltage, current and frequency changes in Czech power plants of similar design after an initiating event which was very similar to the short-circuit in Forsmark. The curves show clearly that no hazard of loss of uninterruptible power supply would follow in case of such an event in Temelín NPP.

Evaluation

According to the present state of knowledge on the Forsmark event, there are no indications of any danger of a scenario such as in Forsmark occurring in Temelín. In particular, there is no hazard of loss of uninterruptible power supply in case of such an event.

As the circumstances of the Forsmark event have not been fully clarified yet, however, the analysis of Temelín experts so far cannot be regarded as final. The work will go on step by step.

Issues of Further Interest

The continuing work of Czech experts regarding the relevance of the Forsmark event for Temelín, which will evolve as more information about the circumstances in Forsmark will become available, should be further monitored.

Accompanying this, the Austrian side should independently follow up the further development regarding the Forsmark event.

10 BROKEN PRE-STRESSING CABLE (NEW ITEM)

(Contributed by VCE)

Introduction

Pre-stressing cables are elements of structural integrity. Therefore any failure of such an element is relevant; in particular in the protective outer hull of the reactor building.

Summary of New Information Provided

This issue was not on the workshop agenda and hence, was dealt with only briefly. Accordingly, there has been no new information provided at the workshop. The acting persons on the Czech side are not experts in the topic and therefore were not able to give a clear picture of the topic and the situation.

The situation is that one out of many pre-stressing cables of the reactor containment cover in Temelín has been found broken during testing. No specific information on type and location is available so far.

Evaluation

Pre-stressing cables are used in concrete structures to control stress situations and structural behavior. They provide redundancy to structures and make them “stronger”. They are composed of a major number of strands which are combined to a cable. Each strand is anchored by wedges on both sides of the cable. These wedges are supposed to be anchored only once in their life-time. Only in exceptional cases such an anchoring should be opened. The testing of the cables by the so called “*lift off tests”* loses the grip of the wedges and introduces damage to the cable (curbs) which can lead subsequently to cable rupture.

The problem of pre-stressing cable failure in reactor buildings is not limited to this single case in Temelín. It seems to be generic for pre-stressed concrete containments, and therefore also for WWER-1000s.

The operators in Kozloduy are currently considering exchanging their cables completely and substituting them by a Western European system (Freyssinet, France). The real cause of the problem can not be properly analyzed due to a lack of information. There is consensus that the Russian system is not very well suited for the executed procedures in the various plants. From Kozloduy it has been reported that anchor plates are breaking brittle and a whole cable has failed recently.

A possible cause for this behavior could be the fact that pre-stressing cables are not made for multiple liftoff testing. The procedure implies that at each testing step a little additional force is introduced into the cable, which consumes the available excess capacity quickly. Furthermore material problems with anchor plates have been experienced several times when non-suitable material has been used. This particularly has been experienced in projects in the Far East, where copy plates from Mainland China have been used instead of the well developed technology of Western European suppliers. Furthermore it has been observed that liftoff tests impose to the bearing plates a completely different stress pattern than the anchoring of the strands. This is consuming lifetime and might be a cause for the breaks experienced.

Liftoff tests are very rarely applied in civil engineering anymore because of the bad experience made. Most of the Western European suppliers insist that not more than one liftoff test shall be performed at each cable and that the anchorage has to be carefully treated to avoid notching of the strands. New monitoring technologies have been developed in order to check cables without the destructive liftoff testing. It should be recommended to apply these technologies also to the nuclear plants.

Issues of Further Interest

It is recommended that the cause for the breaks shall be found out on the generic level, for all plants concerned. After it has been identified it will be necessary to adapt the procedures accordingly. It has to be made sure that the procedures of applying pre-stressing cables and the testing regulations for nuclear power plants are compatible.

With respect to the reported broken cable there are a number of open questions, which are:

- What is the specification of the cable and the anchoring system?
- What is the position of the cable in the structure?
- How many cables are there in which distance?
- What is the implemented inspection procedure?
- What are the functional requirements?

These questions should be followed by the Austrian side. In general the rupture of one cable in such a ridged structure is not an issue of particular interest. Only the question whether this is a systematic problem has to be answered, as similar incidents have been reported from other plants.

The inspection routine should be questioned because it must be avoided to damage a functioning system by inspection.

No preparatory work from the Austrian side is required to address this issue. When the relevant material is presented by the Czech side, an assessment can be carried out within reasonable time.

11 GENERAL IMPRESSIONS FROM WALKDOWN AND WORKSHOP

The programme of the workshop was well-balanced, with the first day aimed mainly at the walkdown in the plant itself and the second day used for presentations of Czech experts and discussion of points of concern (see Appendix 1 for workshop agenda).

The walkdown provided a unique chance to get an overall impression of the plant and its condition. Even critical questions could be discussed. The spirit and open mind of the operator's presentation and explanation shall be particularly mentioned. Obviously the quality of workmanship is excellent.

The following observations have been reported from the walkdown:

- The situation on the 28,8 m stage gave a clear indication that there are major vibration problems. A new damper has been installed at one of the pipes for testing. It has not been mentioned in the presentation on the Safety Case of the Steam and Feedwater lines that this can considerably change the boundary condition of the vibrating system. All the applied technology is standard in industry. No new approaches have been recognized. Permanent vibration monitoring should be recommended in order to detect changes in the system long before a crack can be found by conventional NDT methods.
- From outside it was visible that for the buildings of the emergency generators pre-cast elements have been used. This type of construction is most unfavorable for seismic performance because the elements disassemble, fall down and knock out the machine. For such structures it would be particular interesting to measure the ground characteristic and the predominant structural frequencies in order to assess this scenario.

Compared to the visit in the Dukovany plant 2004 (no reactor walk down) and walk downs in conventional plants Temelín provided the best impression.

Regarding the presentations, the operators of Temelín NPP have presented analyses demonstrating that the plant carefully follows all safety concerns connected with the operational experience. Appropriate conclusions are drawn (presentations on Improvement in A820 Compartment and on Safety Case of the Steam and Feedwater lines; see Appendix 1 for full reference). Temelín experts follow also related events in other NPPs and check Temelín NPP ability to withstand their possible consequences. This has been demonstrated by timely in-depth analysis of recent event in Forsmark, (loss of two trains of uninterrupted power supply due to common cause failure) which was done within the month following the accident and showed Temelín design to be robust and resistant to such hazards (ČEZ safety dept., Letter to SÚJB deputy chairman on nuclear safety, of 25 August 2006, concerning Forsmark-1 event analysis in ČEZ (in Czech)).

A significant amount of information has been provided and many questions were discussed. In the future, however, it might be advisable to avoid too broad an agenda and to focus the program on a smaller number of issues which could then be treated more in-depth.

12 MAIN FINDINGS AND ISSUES OF FURTHER INTEREST

In this section, the most important results of the evaluation of the individual Items are presented. The main issues of further interest which result from the treatment and evaluation of the Items are also listed. Interrelations of Items are identified.

In concluding, it is attempted to identify the Items with the highest priority for follow-up, according to their safety relevance. The urgency of these Items is also assessed.

Highlights from the Items' Evaluation

Regarding the Items treated at the walkdown and workshop, progress generally has been achieved and measures have been taken or are planned for the near future which provides effective improvements.

On the other hand, a number of issues remain open, and there is lack of clarity in several respects.

The most important results of the evaluation of the individual Items are listed here.

28.8 m Level and Related Topics (Item 1)

Vibrations: The current situation (vibration at the acceptable limits in one of the steam lines) would require a very detailed ISI regime. However, the analyses under way are likely to achieve the results needed for an appropriate selection of countermeasures. The work on the bubliks could also close a gap in the qualification of the no-break-zone postulate for the SUPERPIPE application. Resolving the issue should not unduly delayed (also in view of the power uprate envisaged).
The vibration limitation attempts at the High Energy lines are a substantial contribution to the safety of the secondary circuit entering and exiting the containment.

Permanent vibration monitoring should be recommended in order to detect changes in the system long before a crack can be found by conventional NDT methods.

Water Hammer: The changes in the bubliks could make water hammer impact a problem of the past. (This, however, remains to be seen.) Furthermore, new analyses with improved modelling will provide better insights in the nature of the loads, and will allow a limited cross check of postulated break locations. The approach taken seems to be a good start for this issue.

Application of SUPERPIPE concept: Deviations from the basic requirements according to the U.S. ASME code regarding the restrictions on pipe length, the "no branching" requirement and other points remain (see SMR 2.1.4).

Pipelines' fixed points: The fixed points at the transition to the turbine hall have not been improved. They are still not able to withstand loads which might occur in case of pipe breaks (SMR 2.1.4).

Plans for power uprate: At certain occasions, limiting stresses are already reached with extreme load cases based on original nominal power. A concise set of licensing calculations is required for the uprate, going beyond mere updates of the assessments for the current license.

Risk-Informed Inspection: This concept should not be regarded as a patent remedy to reduce the tremendous amount of NDT required at the 28.8 m level. The scope of NDT inspections on the bubliks has not been decided yet. Nozzle corners should be included.

All other areas: It appears that the other shortcomings regarding Item 1 as identified in the SMR (2.1.4) are still valid.

Reactor Pressure Vessel (Item 3)

Preliminary irradiation results: The Czech presentations showed that all results of impact toughness measurements (Charpy tests) were below or close to the embrittlement curve as calculated according to the Russian standards. A possible fluence rate effect of the test reactor irradiation or incorrect irradiation temperatures could have diminished the embrittlement of the specimen, compared to the real RPV material.

ETE irradiation capsules: The first capsules were reported withdrawn in May 2004. Evaluation was to take about one year. Nevertheless, no data were provided at the workshop. They were reported to be available in late 2006/early 2007. The surveillance program is of significant importance for the experimental confirmation of the predicted progress of embrittlement. The most rapid increase in embrittlement is supposed to occur during the first 5 years of irradiation; hence, early results are of particular importance.

Other key areas: Regarding other key areas, the shortcomings as identified in the SMR section 2.3.4 still appear valid.

Integrity of Primary Loop Components (Item 4)

Under-cladding cracks: The question of the NDT capabilities to detect small under-cladding cracks (SMR 2.4) still remains open. Some information on eddy current testing was provided at the workshop, which would need to be corroborated.

Test defects: The doubts that test defects used for qualification of weld inspections do not correspond to the worst case, which were formulated earlier (SMR, 2.4), still remain.

Other key areas: Regarding other aspects of the quality of in-service inspection of main primary loop components, the SMR (2.4) still appears to be valid.

Severe Accidents Related Issues (Item 7b)

Progress of safety upgrading: The information provided generally shows that the work on safety upgrading of the plant is being continued. In some areas the progress is significant. In the area of Severe Accident Management the analyses are continued with improved calculation tools, and are expected to be completed in 2007, after which upgrading of hydrogen recombiners' system is to be expected.

Organization of Severe Accident Management: The corresponding issues have been generally solved and proper training has been assured. There is progress in the development of the design of technical measures needed for severe accident mitigation, and the implementation is under way. However, the process is not finished yet and deserves further monitoring.

Seismic Issues (Items 6 and “Seismic Design”)

New monitoring system: This is a clear improvement of the situation. This topic has been upgraded satisfactory. Still open is the evaluation of the obtained data in order to come to a realistic – measurement based – assessment of the seismic hazard for the Temelín NPP site.

Control Rods and Fuel Degradation (New Item)

Development of Issue: Control rod insertion reliability has been identified as safety issue for WWER-1000/320 NPPs in the early 1990s. It was expected that the problem would not be experienced at Temelín with the new Westinghouse core design. Nevertheless, difficulties did occur in the last years, particularly in Unit 1. A growing number of rod control cluster assemblies failed to achieve full touch-down in the bottom position. Furthermore, at a test on June 02, 2006, two cluster assemblies stopped above the hydraulic dampers and thus failed to meet the Limit Conditions (which have to be fulfilled at all times during operation).

Progress reported: Actions are taken which are based on the experiences in other plants with similar problems. It can be expected that progress will be made.

Leaks from Fuel Rods (New Item)

General assessment of problem: There is no significant deviation from problems as commonly encountered with nuclear fuel cladding failures in many NPPs. It is however worth mentioning, that fuel failures with WWER fuel from suppliers originally in this trade were not increasing to the same degree with the number of service campaigns of the individual FAs. With respect to safety, there is no concern resulting from the quantities of radioactive material released into the primary circuit; they are well below the permissible limit.

Causes of problem: The extended in-service period of the Westinghouse fuel could be part of the problem, but also fretting of the grid-spacers caused by the simple clamping type of fixtures used for holding the fuel rods and the control rod guide tubes in place.

Counter-measures: Removal of the failed FRs and the replacement by solid stainless steel rods appears to be the practice to deal with the problem. This is an adequate procedure and accepted by the licensing authority.

RPV Head Material Degradation (New Item)

Implications for LBB: The operator indicated that already several leakage events have occurred that initiated corrosion at the RPV head. This fact demonstrates that LBB (leak before break) might not be fulfilled, due to inadequate leakage detectability, at least for RPV leaks.

Counter-measures: The cleaning procedures performed were obviously effective; according to the Czech experts NDT analyses have been performed. All reported ISI and maintenance steps are common practice, besides the point that detectability of leakages at the RPV head is evidently not guaranteed by the systems installed at ETE. It therefore is to be expected that aside from a check on feasible improvements of detection, administrative action will be introduced to help avoiding excessive spills, leakages and consecutive corrosion. Generally, it seems that the operator has recognized needs to improve the housekeeping.

Emergency Diesel Generators (New Item)

Consequences of the Forsmark event: According to the present state of knowledge on this event, there are no indications of any danger of a scenario such as in Forsmark occurring in Temelín. In particular, there is no hazard of loss of uninterruptible power supply in case of such an event. As the circumstances of the Forsmark event have not been fully clarified yet, however, the analysis of Temelín experts so far cannot be regarded as final. The work will go on step by step.

Broken Pre-stressing Cable (New Item)

Incident at Temelín: Pre-stressing cables are used in concrete structures to control stress situations and structural behavior. Each cable is composed of a large number of strands which are anchored by wedges on both sides. The testing of the cables loosens the grip of the wedges and introduces damage to the cable (curbs) which can lead subsequently to cable rupture.

One cable has been found broken in Temelín at the occasion of such a test. The topic was not on the workshop agenda and thus has not been sufficiently explained by the Czech side so far.

Generic nature of problem: The problem of pre-stressing cable failure seems to be generic for pre-stressed concrete containments, and therefore also for WWER-1000s. There is consensus that the Russian system is not well suited to the test procedures at the various plants. It could be that the cables are not made for multiple liftoff testing. To the bearing plates, the tests impose a completely different stress pattern than the anchoring of the strands. This is unnecessarily consuming lifetime. In Kozloduy, it is at present under consideration to completely exchange cables and substitute them by a Western European system. Also, new monitoring technologies have been developed to check cable without destructive liftoff testing. These technologies should also be applied to nuclear plants.

Main Issues of Further Interest for all Items

From the lack of information identified for the various Items, and the open issues remaining, the questions which are of further interest and would require further monitoring can be identified.

28.8 m Level and Related Topics (Item 1)

The following issues should be further monitored:

- The current work on the bubliks.
- The vibration limitation attempts at the High Energy lines.
- The further development of NDT applications; particularly in the context of Risk-Informed In-Service-Inspection (RI-ISI).

Regarding other areas which were identified as requiring clarification in the SMR (2.1.4), the SMR still appears to be valid. All those areas therefore should be further observed in the future. In particular, this should be seen in connection with the planned increase of the power output.

No specific preparatory work is required for the Austrian side to permit further discussion of these issues.

Reactor Pressure Vessel (Item 3)

The following issues should be further monitored:

- Development of the embrittlement, with information sufficiently detailed for evaluation.
- Further RPVI relevant issues such as core configuration, RPV internals and loads on those, main coolant line penetrations, vessel head and other penetrations, main flange tightness, coolant chemistry, hydrogen diffusion, corrosion, fatigue, surveillance measures ascertaining LBB applicability, actual RPVI verification and severe accident behaviour (as identified in the SMR, 2.3.4).

No specific preparatory work is required for the Austrian side to permit further discussion of these issues.

Integrity of Primary Loop Components (Item 4)

The following issues should be further monitored:

- NDT-capabilities of detecting small under-clad cracks and differentiating them from cracks within the RPV cladding. Reports and documentation of a complete 4 year inspection cycle would provide the best basis.
- Test defects used for qualification of weld inspections in the primary circuit – particularly regarding the use of worst-case test defects.
- RPV inspection experience.

No specific preparatory work is required for the Austrian side to permit further discussion of these issues.

Severe Accidents Related Issues (Item 7b)

The completion of severe accident analyses, expected in 2007, should be further monitored.

Also, the further development regarding the design of technical measures deserves monitoring. In particular, this concerns the following points:

- Upgrading of hydrogen recombiners.
- Measures for enlargement of the molten core area:
  - Outer reactor cavity door (hermetic door) – provisions for opening, or inserts which melt easily are envisaged
  - Removable barriers to protect containment wall from corium
- Stuffing of ex-core ionization chambers' channels (to be implemented 2007).
- Enlargement of coolant inventory inside containment.

No specific preparatory work is required for the Austrian side to permit further discussion of these issues. The final, implemented solution should be thoroughly reviewed and verified against calculations concerning the velocity of melt-through with the enlarged area.

Seismic Issues (Items 6 and “Seismic Design)

Only one specific topic has been touched in detail in this meeting, leaving all the other topics of the SMR and FMR open for further bilateral contacts and clarification. The recommendations brought forward in the Austrian workshop presentation are in line with the recommendations of the SMR and were meant to improve the understanding of these issues.

- When the seismic issue will be opened in the course of the 10 year safety review, it should be checked whether the new recommendations have been adopted and implemented in the promised probabilistic seismic hazard assessment to be performed.
- Furthermore, it would be of interest to receive raw data of the monitoring system in order to carry out an assessment beyond the scope of the normal operation of the system. This assessment could bring valuable information on regional seismic properties such as seismic impedance and wave speeds.

The Czech colleagues from the University of Brno have showed interest in this collaboration and promised the submission of data in case that the plant management agrees. At the meeting there has been no objection from the plant operator.

Control Rods and Fuel Degradation (New Item)

Due to the potentially high relevance of this topic, there is a number of questions which should be followed up with urgency.

A part of these questions relate to the five requirements listed in the IAEA’s “Issue Book” on WWER-1000/320s (1996), concerning:

- Operational counter-measures (operation at reduced power)
- Drop times and drop tests
- Fuel loading and burn-up strategies
- Tests of lifting an lowering forces
- Structural counter-measures (readjustment of upper internal structure)

Other questions considered as relevant go beyond the IAEA requirements:

- Investigations concerning safe shutdown in accident situations
- Fuel replacement strategies (from Westinghouse to TVEL fuel)
- State of knowledge regarding root causes
- Consequences of test loads for components’ lifetimes

Leaks from Fuel Rods (New Item)

There is no immediate safety concern arising from this Item.

- Considering the safety relevance to be attributed to the effects of leakage in case of an accident, the number of leaks gives good reason to further observe the development of the leakage rates.
- The fuel is supposed to be changed. This should also have an impact, reducing the incidence rate of failures, and should be observed.

No specific preparatory work is required for the Austrian side to permit further discussion of these issues.

RPV Head Material Degradation (New Item)

The following issues should be further monitored:

- Improvements of leak-detection capability in the primary circuit, particularly the RPV.
- Administrative measures which will be introduced to help avoiding excessive spills and leakages.

No specific preparatory work is required for the Austrian side to permit further discussion of these issues.

Emergency Diesel Generators (New Item)

- The continuing work of Czech experts regarding the relevance of the Forsmark event for Temelín, which will evolve as more information about the circumstances in Forsmark will become available, should be further monitored.
- Accompanying this, the Austrian side should independently follow up the further development regarding the Forsmark event.

Broken Pre-stressing Cable (New Item)

The cause for the breaks should be found out on a generic level. After it has been found, the test procedures have to be adapted accordingly.

With respect to the reported broken cable there are a number of open questions, which are:

- What is the specification of the cable and the anchoring system?
- What is the position of the cable in the structure?
- How many cables are there in which distance?
- What is the implemented inspection procedure?
- What are the functional requirements?

These questions should be followed by the Austrian side. In general the rupture of one cable in such a rigid structure is not an issue of interest. Only the question whether this is a systematic problem has to be answered, as similar incidents have been reported from other plants.

The inspection routine should be questioned because it must be avoided to damage a functioning system by inspection.

No preparatory work from the Austrian side is required to address this issue. When the relevant material is presented by the Czech side, an assessment can be carried out within reasonable time.

Interrelation of Items

Of the Items treated here, the Item “Integrity of Primary Loop Components” is clearly and closely related to the RPV Item, particularly regarding the detectability of under-cladding cracks, which provides an important input to PTS analyses. This interrelation has already been identified and should be kept in mind in the further treatment of the Items.

Of the new Items, the Item of “Control Rods and Fuel Degradation” is related to the “Leaks from Fuel Rods”, since both involve fuel behavior and fuel problems, if seen from a different angle. Counter-measures, particularly changes in the fuel, triggered by one of those Items could well have consequences for the other.

Priorities for Follow-up

The following Items appear to be the most important ones for follow-up, according to their safety significance. Urgency of follow-up is high in the first two cases, medium to high in the third.

Control Rods and Fuel Degradation (New Item)

This is a new problem. Control rods are parts of one of the most crucial safety systems of an NPP. There are a number of questions which should be followed up with urgency; measures are planned but also need to be followed up.

High priority, high urgency.

28,8 m Level and Related Topics (Roadmap Item 1)

Still unresolved safety issues persist (application of SUPERPIPE, water hammer impact, break locations… see SMR 2.1.4). At the same time there are acute vibration problems requiring modifications, against the background of an envisaged power uprate.

High priority, high urgency.

RPV and Primary Circuit (Roadmap Items 3 and 4)

So far, no surveillance results have been provided; the first results should be available in the near future. Other issues are still open as well (PTSA, application of VERLIFE, detection of small under-cladding cracks… see SMR 2.3.4)

High priority, medium to high urgency (embrittlement progresses only gradually; on the other hand, most of the embrittlement occurs in the first 5 years of operation).

ABBREVIATIONS

AC...................... alternating current

AM .... accident management

ASME................. American Society of Mechanical Engineers

BDBA .... beyond design basis accident

BRUA................. main steam relief valves

ČEZ................... electrical utility, owner of Temelín NPP

DC...................... direct current

ECR .... emergency control room

ENCO .... ENCONET Ges.m.b.H., Vienna

EOP .... emergency operating procedure(s)

ETE.................... elektrarna Temelín (power plant Temelín)

FA .... fuel assembly

FMR .... final monitoring report

FR .... fuel rod

HVAC................. heating, ventilation and air conditioning

IAEA................... International Atomic Energy Agency

IAM .... Institute of Applied Mechanics, Brno

IEC..................... International Electrotechnical Commission

IRR..................... Institute of Risk Research, Vienna University

I&C..................... instrumentation and control

ISI .... in-service inspection

LBB .... leak before break

LOCA .... loss-of-coolant accident

MCCI.................. molten core – concrete interaction

MCR................... main control room

MSSL .... main secondary steam line

MSSSV .... main secondary steam safety valves

NDT .... non-destructive testing

NPP .... nuclear power plant

NRC .... Nuclear Regulatory Commission (USA)

NRI..................... Nuclear Research Institute, Řež

OLC .... operational limit condition

PAMS................. post accident monitoring system

PAR .... passive autocatalytic recombiner

PN...................... project number

PORV................. pilot operated relief valve

PSA.................... probabilistic safety analysis

PSHA .... probabilistic seismic hazard assessment

PTS.................... pressurized thermal shock

PTSA .... pressurized thermal shock analysis

QA...................... quality assurance

RCCA................. rod control cluster assembly

RI-ISI.................. risk-informed in-service inspection

RPV .... reactor pressure vessel

RPVI .... reactor pressure vessel integrity

SA .... severe accident

SAFT.................. synthetic aperture focusing technique

SAG .... severe accident guideline(s)

SAM .... severe accident management

SAMG .... severe accident management guideline(s)

SCG .... severe challenge guideline(s)

SG...................... steam generator

SGTR .... steam generator tube rupture

SÚJB.................. Czech State Office for Nuclear Safety

TOFD .... time-of-flight-diffraction

TSC.................... technical support center

TSO .... technical support organisation

TVEL.................. Russian state-owned nuclear fuel manufacturer

USNRC .... U.S. Nuclear Regulatory Commission

UT .... ultrasound testing

VCE .... Vienna Consulting Engineers Holding GmbH

VERLIFE............ Unified Procedure for Lifetime Assessment of Components and Piping in WWER NPPs

WWER .... Water-Water-Energy-Reactor, reactor type developed in the Soviet Union

APPENDICES

Appendix 1:

1. Agenda of the Workshop 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, Temelín, September 26 and 27, 2006
2. Listing of Presentations provided by the Czech side at this workshop

Appendix 2:

Detailed Report on the Workshop in Temelín, September 26 and 27, 2006, by the Institute of Risk Research (IRR), Vienna University
Confidential – Seperate Volume

Appendix 3:

Detailed Report on the Workshop in Temelín, September 26 and 27, 2006, by Enconet Consulting Ges.m.b.H. (ENCO), Vienna

Confidential – Seperate Volume

Appendix 4:

Seismic Monitoring and Assessment – Results of the Evaluation and Findings; Presentation by H. Wenzel, Vienna Consulting Engineers (VCE), at the Workshop in Temelín, September 26 and 27, 2006

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