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1st Nuclear Knowledge Preservation & Consolidation (NKP&C) Workshop WWER - WS1

Summary

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1st Nuclear Knowledge Preservation & Consolidation (NKP&C) Workshop

WWER - WS1

Amsterdam, 11-13 December 2007

Summary Record

1 Background

Nuclear knowledge had been build up continuously since the middle of the last century. After Chernobyl in 1986 the public opinion changed leading to a gradual phasing out process of nuclear energy in several Member States. During that time a trend at universities and in industry was observed of a decrease in students choosing nuclear related studies. Now the generation of senior nuclear experts is retiring. On the other hand, due to security supply and climate change issues (green house mitigation measures) receiving more importance lately, a renaissance of nuclear power is ongoing. In order to avoid a possible loss of capability and knowledge in the EU action is taken now preserving and disseminating it to the new generation.

There is a huge amount of information and knowledge available, either published or easily available, but also publications difficult to trace. Especially those are at risk of being dispersed or lost due to a series of factors, including:

- retirement of Senior Experts who were present at the time when most Nuclear Power Plants were designed and put into operation,
- generational gap (due to years of decline in new constructions, only a limited number of people started their career in that area)
- non-electronic publishing in the past
- limited dissemination possibilities
- language (many non-English publications from Eastern countries)

Therefore, the Institute for Energy of the Directorate General Joint Research Centre has developed a method for consolidation of nuclear knowledge (fig. 1).

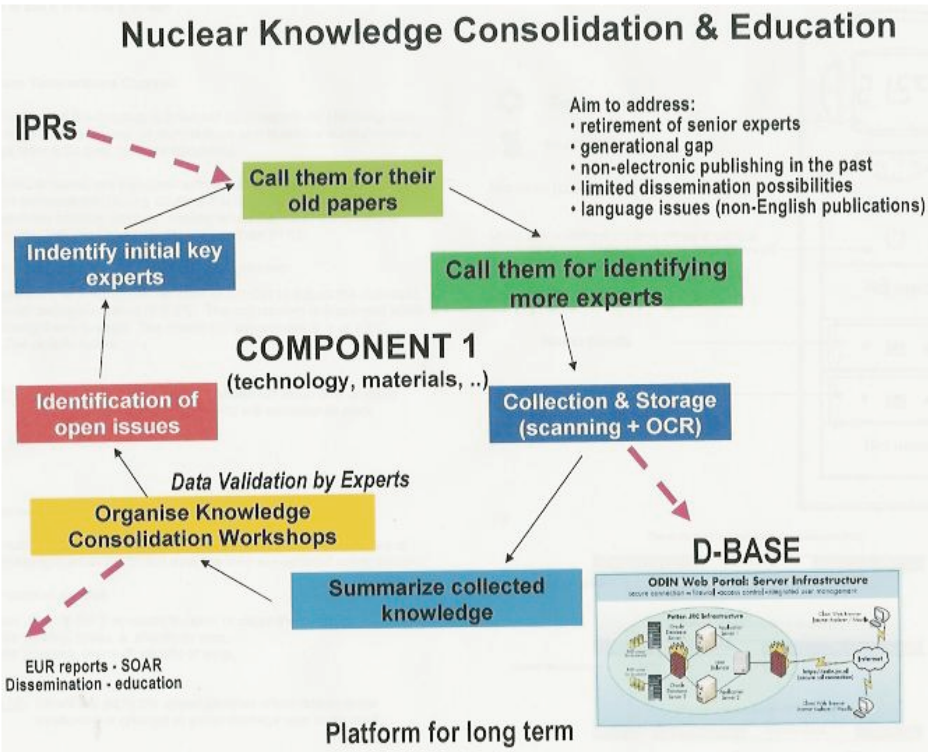

Figure 1: Nuclear Knowledge Consolidation Method

The method relays on the mobilisation of all identified leading experts in the EU in re-evaluating old knowledge and consolidating what is necessary to create training materials for the new generations.

The methodology is applied for the present pilot study for consolidating and preserving WWER RPV safety related literature, which could be the start of a wider Nuclear Knowledge Preservation and Consolidation activity in the Nuclear Design Safety unit of the Institute for Energy.

2 Scope

Several reviewers received between 7 and 21 papers in their field of expertise, in order to review the content and present it for discussion and consolidation to the WWER Reactor Pressure Vessel embrittlement experts during the workshop.

The reports and presentations were requested to follow the below structure:

- per paper
  - paper title, author(s), reference
  - reviewers summary/abstract
  - comments on "up-to-date-ness" of the papers/material
- conclusion on the complete review:
  - more reference papers in the area known to the reviewer
  - open issues in the area known to the reviewer

The short-term scope is to reach a consolidated conclusion for the individual reviews, after the discussion and consolidation process during the workshop. The medium-term scope is a consolidated review in the individual expert fields. The long-term scope is to prepare a State-of-the-Art report for the complete WWER RPV Irradiation Embrittlement expert field, incl. the history and reasons of the choices made (material, composition, etc.). The last general document was produced more than 26 years ago by Nikolaev, Amaev and Alechenko, which is in Russian and needs upgrading.

In the brainstorming session of the workshop the predefined fields of expertise in WWER RPV Embrittlement were discussed and redefined as described in figure 2.

New sub-fields		***Old sub-fields***
1.	SOL toughness	*%*
2.	Irradiation shift prediction	*Modelling – T. Williams*
3.	Property-property correlation	*Testing/FT – F. Gillemot/A. Ballesteros*
4.	Annealing and re-irradiation	*Annealing – A. Chernobaeva*
5.	Material Factors	*Chemistry – A. Kryukov*
6.	Environmental factors	*% and dosimetry*
7.	Mechanism & micro structural evolution	*Microstructure – V. Slugen*
8.	PLEX Issues	*PLEX – R. Ahlstrand*
9.	Surveillance	*Surveillance – L. Kupca*
10.	Cladding	*%*

Figure 2: Subdivision of WWER RPV Embrittlement Expert Fields

In the below chapters the reviews per expert field (in brackets: the new denominations) are summarized.

3 Chemistry (Material Factors)

Session Summary

Six of the seven reviewed papers were devoted to WWER RPV steels. One is the comparison of Eastern and Western RPV irradiation embrittlement. The correlation between mechanical properties and micro structural results is performed in four papers. Results of RPV material irradiation embrittlement modeling are presented in 1 paper. Mechanical testing data analysis is presented in two papers.

Consolidated Conclusions

For WWER-440 RPV steel the main influencing elements are Cu and P. The Ni content is less then 0.3 %.

The Cu effect due to neutron irradiation is:

- significant level of Cu depletion up to 0.03- 0.04% in irradiated weld ferrite matrix
- ultrafine (2-3 nm) intra-granular clusters enriched in - Cu, P, Ni, Mn and Si
- cluster density and Cu content in cluster increases as fluence increases

The Cu and flux relation is:

- Flux effect (Tk shift) depends on Cu content. If the Cu content is higher than ~0.13%, the flux effect could be significant
- Cu contribution (plateau) does not depend on flux value
- Cu hardening kinetic depends on flux
- Cu hardening goes on faster with low flux

The Phosphorus effect is:

- significant level of P depletion (up to 0.003-0.004% in irradiated weld ferrite matrix
- ultrafine (2-3 nm) intra-granular clusters enriched in - Cu, P, Ni, Mn and Si
- intragranular P atmospheres detected in irradiated materials

The Phosphorus influence mechanisms are:

- -intergranular segregation (non-hardening embrittlement)
- -segregation on Cu precipitates
- -phosphorus atmospheres and clusters
- -segregation on dislocations

For WWER-1000 RPV steel the main influencing element is nickel. Nickel content is between 1,0 and 1,9 %. Phosphorus and cupper contents are low. The mechanism of Ni influence is still unclear.

There is no Ni influence in Russian Guides for WWER-1000 RPV irradiation embrittlement assessment.

Open Issues

In this field of expertise the unique data and knowledge should be preserved. There is presently the possibility existing for different analyses, such as mechanical testing & micro structural research, Western & Eastern data, model & commercial steels, high & low fluxes, test reactor & surveillance data, etc.

The general open issues are:

- Basic item – incomplete information (e.g., an example is flux)
- Only papers from magazines, not from conferences, seminars, workshops

The open issues for WWER-1000 RPV material are:

- Has Ni influence to matrix damage?
- Ni-Mn synergism

- Role of flux

Reviewed papers and summaries

- *Effect of nickel, manganese and copper on irradiation sensitivity of alloy steel welds - T. Williams, Pressure vessel and piping 81 (2004) 657-665*

The similarities and differences between "Eastern" and "Western" steels are explored by comparison between published data on WWER steels, and data and models developed on Western type welds with similar nickel contents to the WWER-440 and WWER-1000 steels. The effects of nickel, manganese and copper on the Charpy specimens' transition temperature shift and hardness due to irradiation had been analyzed. The results of extensive program of micro structural examinations (SANS and AP) were also used. The paper concludes with a proposal for further investigation, i.e. to develop the model for all RPV assessment by the same set of equations.

- *Role of nickel in a semi-mechanistic analytical model for radiation embrittlement of model alloys - L.Debarberis, B.Acosta, F.Sevini, A.Kryukov, F.Gillemot, M.Valo, V. Nikolaev, M.Brumovsky, Journal of nuclear materials, 336 (2005), 210-216*

The improvement of a semi-mechanistic model for transition temperature assessment by taking into account Ni content is proposed. The comparison of residuals by fitting models with the Ni contribution and without Ni contribution is presented.

- *Irradiation embrittlement of model alloys and commercial steels: analysis of similitude behaviors - L. Debarberis, F. Sevini, B.Acosta, A.Kryukov, Y.Nikolaev, A.Amaev, M.Valo, Pressure vessel and piping, 79 (2002) 637-642*

A parametric study of the response to neutron irradiation of 32 different model alloys with parametric variation of Ni, P and Cu has been carried out. To demonstrate the usefulness of the study to commercial RPV steels, an analysis of the results and the similitude of behavior between model alloys and RPV commercial steels has been carried out and the results are presented in this paper.

- *Radiation embrittlement of low-alloys steels - Y. Nikolaev, A. Nikolaeva, Y. Shtrombakh, Pressure vessel and piping 79 (2002)*

The results of phosphorus, copper and nickel effect on radiation induced yield stress increase and DBTT shift are presented. Basic results of WWER-440 and WWER-1000 surveillance and research programs are discussed. The basic regularities of WWER-440 and WWER-1000 RPV steels are discussed. The annealing effectiveness for WWER-440 and WWER-1000 steel was compared. DBDT recovery of WWER-10000 was found to be much lower than for WWER-440 steels. Nickel was supposed to increase the post-irradiation residual DBTT shift of WWER-1000 type steels.

- *Composition effects on the radiation embrittlement of iron alloys. J. Bohmert, J. Ulbricht, A. Kryukov, Y. Nikolaev, D. Erak, ASTM STR 1405, (2001)*

The effects of copper, phosphorus and nickel on radiation damage of model alloys were investigated both mechanical testing (Charpy specimens) and small angle neutron scattering (SANS) method. The Charpy testing results show that DBTT shift increases with Cu, P and Ni content.

- *Irradiation effect on toughness behavior and microstructure of WWER-type pressure vessels - J. Bohmert, H.-W. Viehrig, A. Ulbricht, Journal of nuclear materials, 297 (2001) 251-261*

The results of WWER materials irradiated in "Reinsberg" NPP with the temperature 255 C are presented. Mechanical testing (Charpy specimens) and microstructure investigations (small angle neutron scattering –SANS) have been performed. There is an obvious correlation between DBTT shift after irradiation and irradiation-induced changes in microstructure.

- *Embrittlement of low copper WWER-440 surveillance samples neutron-irradiated to high fluences - M.K. Miller, K.F. Russel, J. Kocik, E. Keilova, Journal of nuclear materials 282 (2000) 83-88*

The paper is devoted to atom probe tomography microstructure investigation of WWER RPV material, irradiated by fluence 3-5 times more then WWER-440 end of life one. The basic conclusion of analysis – changes in mechanical properties correlate with the presence of manganese-, silicon-,copper-,phosphorus- and carbon-decorated dislocations and other features in the matrix of irradiated materials.

4 Annealing (Annealing and re-irradiation)

Session Summary

21 papers were reviewed in this expert field. Most papers were encountered in the mid '90s, when annealing was an important issue (fig. 3).

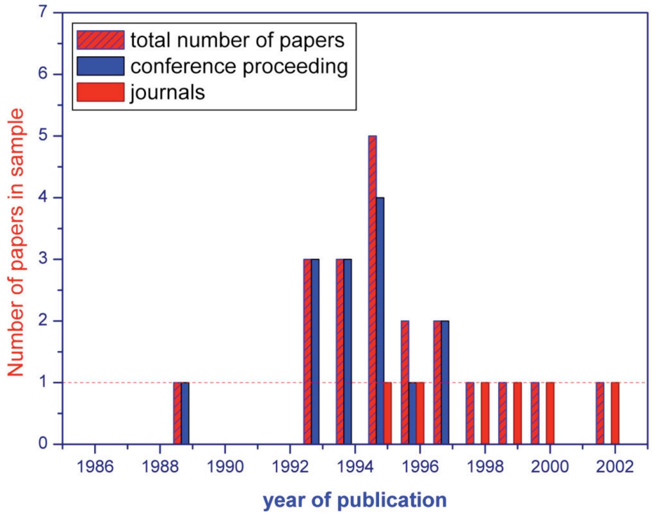

Figure 3: Distribution of reviewed papers by year of publication

The Ni content of the steels varied between 0.05-2.45%wt. The Mn content of the steels varied between 0.44-1.61%wt. The P content varied between 0.005-0.055%wt. The Cu content of the steels varied between 0.03-0.46%wt.

The following problems seem to be important for the RPV annealing:

- The study of the mechanical properties recovery due to annealing
  - Temperature

  Amaev et al. (1993, 1994) report that the effect of temperature on recovery of properties of WWER-440 steels has been studied using Charpy testing and that the temperature has to be above 420°C. Nanstad et al. (1997) conclude that annealing at 343°C leads to almost full recovery of USE and 50% recovery of T41J of US weld and base materials. Almost full recovery of US and Russian steels was reached after annealing at 454°C/168h. Nikolaev et al. (1995) conclude, that after 72 annealing the residual embrittlement of WWER-1000 materials depends on temperature, with the best efficiency at 460°C and 490°C. Vacek et al. (1993) report that after 1 hour annealing of WWER-1000 type material the recovery of $R_{p0.2}$ increases when T_{ann} increases from 300 up to ~500°C, but does not reach full recovery even at 600°C. The increase of the temperature 450-600°C (1 h) does not effect the recovery efficiency. Kohopää et al. (2000) state that the effectiveness of annealing is following the line 400°C-450°C-475°C/100h.

  - Annealing time

Amaev et al. (1993, 1994) show that the most intense reduction of micro hardness occurs during the first hours. Then process slows down and stabilizes at the level, characteristic of the given T_{ann}. Vacek et al. (1993) demonstrate increasing recovery of $R_{p0.2}$ from 50% after 1 hour up to ~100% recovery after 50 hours annealing time for WWER-1000 type material at 425°C. Popp et al. (1989) conclude that 24 hours annealing is enough for the effective recovery of WWER -440 materials at 425°C and 450°C.

- Chemical compositing

Amaev et al. (1993, 1994) conclude that higher phosphorus content creates higher residual embrittlement. This effect is most evident at 340°C and 420°C and less at 460°C. Vacek et al.(1993) state that recovery of HV depends on the Cu content. Nikolaev et al. (1995) show that residual embrittlement of WWER-1000 steels does not depend on the chemical composition if Ni≤1.74% after annealing at 460°C/72 h.

- Fluence

Amaev et al. (1993, 1994) encounter no effect of fluence on T_K(residual). Bumovsky et al. (1995) note just a slight fluence effect. Nikolaev et al. (1995) had results showing tat there is an effect of fluence on residual embrittlement of WWER-1000 steels. The higher the fluence the higher is the residual Embrittlement.

- The development of the annealing regime

No information or discussion about the way how to use the results of experiments to receive parameters for annealing of RPVs

- The validation of the annealing efficiency

  - Residual embrittlement (ΔT_K(res)) assessment

Amaev et al. (1993, 1994) showed that after annealing of WWER-440 steels at 460°C/100h the residual embrittlement is not higher then 40°C, whereas the medium value is 20°C. Brumovsky et al. (1995) demonstrate that after annealing of WWER-440 steels at 475°C/168h the residual embrittlement is 5 - 46°C, whereas the medium value is 30°C. Kohopää et al. (2000) presented results where the residual embrittlement after annealing of WWER-440 steels at 475°C/100h is 14-18°C. Lucon et al. (2002) showed that the residual embrittlement after annealing of WWER-440 steels at 475°C/100h is 1°C. Valo et al. (1997) concluded that the residual embrittlement after annealing of WWER-440 steels at 475°C/100h is 25°C.

  - Response of different parameters (T_K, USE, T_0, $R_{p0.2}$, HV) on annealing

Brumovsky et al. (1995) concluded that after annealing of WWER-440 steels at 475°C/168h the residual $R_{p0.2}$, is about 50 MPa. There is an over recovery of USE. T_0residual is about 20°C. Kohopää et al.(2000) found that after annealing of WWER-440 steels at 475°C/100h residual T_0 is 7-17°C with full recovery of USE. Lucon et al. (2002) stated that after annealing of WWER-440 steels at 475°C/100h there is almost full recovery of $R_{p0.2}$ and over recovery of USE with T_0residual of about 12°C. Nanstad et al. (1997) reported that after annealing of Russian and US type of steels at 454°C/168h there is over recovery of USE. Nikolaev et al. (1995) concluded that annealing at 400°C leads to full recovery of $R_{p0.2}$, but not to full recovery of T_K.

- The technical problems concerning the annealing

Brumovsky et al. summarized that the old type of WWER-440 RPVs have several features, i.e. a relatively small diameter of vessel, which means that the EOL fluence is high ~2×1020 cm-2; not enough information about materials, as en example Cu and P content and T_K0; a high rate of embrittlement; no surveillance programs; annealing was chosen as a way of mitigation; for the uncladded vessels it is possible to take some materials from inner surface; for the cladded vessels it is possible to take some materials from outer surface. Ahlstrand et al. concluded that the annealing regime is 475°C/100h; the heating and cooling rate is ≤20°C/h; the total annealing time is 9-11 days; the temperature of the weld and upper and low pars and concrete has to be measured by several thermocouples; it is necessary to make NDT inspection after annealing; the main problem for Loviisa is the impossibility to test real specimens from the RPV. Brynda et al. report that the annealing temperature should be in a window high enough for recovery of the properties, but lower than a temperature, which can impact the other components, such as primary piping, support structure and biological shielding, concrete, etc. There is also concern mentioned about the temper embrittlement during annealing. It is necessary to control the temperature during annealing at several points.

- Licensing problems for the annealing

Brumovsky et al. mentioned that the wide supporting program had been performed by Scoda and NRI for annealing. The program included the design of the furnace; the calculation of thermal and stress fields in reactor; the decision to anneal weld № 4 and the nearest aria; the decision to add an additional ring to the furnace to decrease the temperature gradient; a special demonstration experiment which had to be done with full size vessel dimensions; a program of irradiating and annealing specimens; a cutting of a template form of the outer surface in order to check the chemical composition. Ahlstrand et al. reported that studies of annealing had been started in 1980. Later, a systematic research program from Loviisa surveillance specimens was performed. A separate part of the licensing program was carried out by Moch-Otshig in Russia. This program included a study for the cladding.

- Re-irradiation behaviour

Amaev et al. (1993) state that the rate of the embrittlement depends on the residual embrittlement after annealing. Three models of re-irradiation were proposed: lateral shift, conservative and vertical models. The lateral shift is presented as the most adequate model. Amaev et al. (1994) state also that the re-embrittlement depends on the phosphorus content. Kohopää et al. (2000) propose a new re-embrittlement model. Lucon et al. (2002) found out that the rate of re-embrittlement is lower then rate of embrittlement. Nanstad et al. (1997) concluded that the rate of re-embrittlement is following lateral shift for US and Russian steels.

- Surveillance program for the post annealing operation

Ahlstrand (1995) reported about a new surveillance program for monitoring of re-irradiation of weld № 4 consisting of 300 specimens.

- Assessment of the RPV state

The subject can be subdivided into two sub-fields:

- Comparison between properties of material, cut form RPV with specimens irradiated in surveillance channels and research reactors
- Correlation between standard and sub size specimens

Kohopää et al. (1994) investigated small samples from the outer wall at the core region of Loviisa 1, which were cut out in 1993 from 2 forgings above and below weld № 4. The 3×4 Charpy specimens were tested and used for correlation to convert the data to standard Charpy test. Mechanical tests, chemical analyses, fluence determination and microstructure study were performed. It was concluded, that surveillance specimens represent base metal well. Further results were that the lifetime of Loviisa-1 is not limited by base metal irradiation embrittlement. Ahlstrand et al. (1995) reported that small samples from the inner wall at the core region of NPP Greifswald 2 were cut out before and after annealing. The 3×4 Charpy specimens were tested by chemical analyses, fluence determination and micro-hardness. It was concluded, that the correlation between sub size and standard Charpy specimens need improvement and confirmation. Kryukov et al. (1995) reported that small samples from the inner wall at the core region of Kozloduy 2 were cut out in 1992 from the weld №4 and the forging after 16 years of operation. The 3×4 and 5×5 Charpy specimens were tested. The chemical composition was obtained. The actual TKs after irradiation and post-irradiation annealing were determined and the reconstruction of TK_0 was assessed. A correlation between sub-size and standard Charpy specimens testing was determined. Valo et al. (1995) obtained results from 3×4 Charpy specimens from Greifswald 2, which were compared with surveillance data of Loviisa 1. The data from Greifswald 2 does not point directly on a dose rate effect, but it does not exclude it. It was concluded that fracture toughness behavior should be checked in the surveillance program. Valo et al. (1998?) introduced a technique for manufacturing of mini Charpy specimens instead of standard Charpy specimens. The VTT type of correlation between 3×4 and standard Charpy specimens was proposed. It was concluded that it is possible to use mini-Charpy specimens for the monitoring of transition temperature shift. Lucon et al. (2002) described how Standard and 3×4 Charpy specimens from a WWER-440 weld were irradiated, annealed and re-irradiated in a research reactor. A correlation between sub-size and standard Charpy specimens testing was analyzed. It was concluded that with sub-size specimens in comparison with standard Charpy specimens testing TK was underestimated.

Consolidated Conclusions

The following general conclusions can be drawn from the review:

- Research results presented at conferences are very important, because the new information comes faster than through journals
- This review summarizes the main questions, discussed in this field of expertise
- This review has been performed on the basis of the conclusions made in the papers.
- The next step should be the data analysis of those referred to in the papers.

Open Issues

The general open issues are:

- Re-embrittlement modelling
- Micro structural mechanisms
- Ni role in RPV annealing

Reviewed papers and summaries

- *Annealing of the reactor pressure vessel of the 1st unit in Loviisa nuclear power station – R. Alstrand, IAEA specialists meeting, Epso, Finland, (1995)*
- *Evaluation of material properties of the reactor pressure vessel in NP S Greifswald Unit 2, before and after annealing – R. Alstrand, M. Valo, Yu. Kohopää, international workshop on WWER-440 reactor pressure vessel embrittlement and annealing (1994) Slovakia*
- *Mitigation of radiation damage by annealing – A. Amaev, A. Kryukov, M. Sokolov, Int.Symposium FONTEVRAUD III, Volume 2,12-16 Sept.1994, 602*
- *Recovery of the transition temperature of irradiated WWER-440 vessel metal by annealing – A. Amaev, A. Kryukov, M. Sokolov, Radiation embrittlement of nuclear reactor pressure vessel steels, An international review (Fourth volume), ASTM STP, p.369-379, 1170,1993, USA*

- *Pressure vessels materials – M. Brumovsky, M. Kytka, T. Pav, P. Novosad, IAEA specialists meeting, Epso, 1995*
- *Reactor pressure vessel annealing- effective mitigation method - M. Brumovsky, J. Brynda, International Conference on Nuclear Engineering, volume 5, ASME,1996*
- *Annealing process temperature and recovery of mechanical properties of weakened RPV materials – J. Brynda, IAEA specialists meeting, Epso, 1995*
- *Exploratory study of irradiation, annealing and irradiation effects on American and Russian reactor pressure vessel steels – A. Chernobaeva, M. Sokolov, R. Nanstad, A. Kryukov, Yu. Nikolaev, Yu. Korolev, Proceeding of the eighth international symposium on Environtmrntal degradation of materials in nuclear power systems – water reactors, volume 2 Florida, USA 1997*
- *The principals structural changes proceeding in Russian pressure vessel steels as a result of neutron irradiation, recovery annealing and re-irradiation – B. Gurovich, E. Kuleshova, O. Lavenchuk, K. Prikhodko, Ya. Shtrombakh, Journal of nuclear materials, 264 (1999) 333-353*
- *Re-embrittlement behavior of WWER-440 reactor pressure vessel weld material after annealing, - Yu. Kohopää, R. Alstrand,, International Journal of pressure vessel and piping, 77 (2000) 575-584*
- *Evaluation of the radiation damage of the reactor pressure vessel at Loviisa 1 using samples taken from the outer surface – Yu. Kohopää, R. Alstrand, M. Valo, International workshop on WWER-440 reactor pressure vessel embrittlement and annealing 1994, Slovakia*
- *Effects of Post-Irradiation Thermal Annealing on Kadiation Embrittlement Behaviour of Cr-Mo-V Alloyed Weld Metals- Yu. Kohopää, Acta politechnika, Mechanikal ingineering service, 132, Epso Finland 1998*
- *Investigation of samples taken from Kozloduy unit 2 reactor pressure vessel – A. Kryukov, P.Platonov, Ya. Shtrombakh, V. Nikolaev, E. Klaustnitzer, C. Leitz, C-Y Rieg, Nuclear engineering and design, 160 (1996) 59-78*
- *SCK-CEN contnbution to the IAEA Round Robin exercise of WWER-440 RPV weld material: irradiation, annealing, and re-embrittlement - E. Lucon, E. van Valle, M. Skibetta, R. Chaouadi, M. Willikend, M. Weber, International Journal of pressure vessel and piping, 79 920020 665-684*
- *Effects of thermal annealing and reinadiation on toughness of reactor pressure vessel steels – R. Nanstad, S. Iskander, M. Sokolov, A. Chemobaeva, Yu. Nikolaev, A. Kryukov, Yu. Korolev, n/d 1997*
- *Radiation embrittlement and thermal annealing behavior of Cr-Ni-Mo reactor pressure vessel materials – Yu. Nikolaev, A. Nikolaeva, A, Kryukov, V. Levit, Yu. Korolev, Journal of nuclear materials 226 (1995)144-155*
- *Radiation Embrittlement and annealing recovery of CrNiMoV pressure vessel steels with different copper and phosphorus content – M. Vacek, P. Novosad, R. Havel, Radiation embrittlement of nuclear reactor pressure vessel and steels, An international review (Fourth volume) ASTM STP,p. 172-182, 1170,1993, USA*
- *Annealing response of Greifswald Unit 2 core weld measured wit pressure vessel samples – M. Valo, R. Ahlstrand, IAEA specialists meeting, Paris, France, 1993*
- *Annealing behavior of Loviisa 1 surveillance materials measured with V-notched specimens and the currents status of vessel anneal – M. Valo, Yu. Kohopää, K. Wallin, T. Planman, n/d 1997*

- *Investigation of irradiation embrittlement and annealing behavior of JRQ pressure vessel steels by instrumented impact tests – Valo M., Rintamaa R., Nevalainen M., Wallin K., Torronen K., Tipping P., IAEA specialists meeting, Paris France, 1993*
- *Evaluation of thermal annealing behavior of neutron –irradiated reactor pressure vessel steels using non destructive test methods – K. Popp, G. Brauer, W-D. Lconhardt, H.W. Viehrig, Radiation embrittlement of nuclear reactor pressure vessel steels. An international review (Fourth volume) ASTM STP, p. 188-205, 1011, 1998, USA*

5 Modelling (Irradiation Shift Prediction)

Session Summary

This chapter reviews/summarizes eleven papers from the "Modelling" folder in the ODIN database. They can be loosely sub-categorized as follows (the figure in parentheses is the number of papers in the sub-category):

- Irradiation damage modelling (6)
- Mechanisms (1)
- Fracture toughness modelling (4)

In all sub-categories, the papers reviewed represent only a small part of the tip of an iceberg. Obviously also they represent European work and WWER interests, whereas much non-European, non-WWER, work may be of direct relevance to these interests. Some of the papers are now quite old and may have been overtaken by further developments in experimental data and understanding. To attempt a detailed review of these papers in the context of current world knowledge would be a formidable task; necessarily the following will fall a long way short of this.

The irradiation modelling papers all, to a greater or lesser degree, recognize the importance of a firm physical basis for model construction. This was particularly powerful in those dealing with re-irradiation after annealing where understanding can provide the confidence to take advantage of good experimental results. The papers also showed a willingness to innovate and depart from convention where appropriate. Usually, the models were supported by some experimental data, though a general weakness in this respect was the amount of data and a general lack of microstructural data to support some of the assumptions.

Consolidated Conclusions

The present way of modelling is by fitting mechanistically-guided equations to empirical data.

For the future, multiscale multiphysics models, such as those being developed in the PERFECT Project may provide more exact predictions, and may be reliably extrapolable outside the range of the data.

However, such models cannot currently represent the complexities of steels and are limited to predictions for simple iron-copper alloys, though nickel and manganese effects may soon be added, and phosphorus later. It may be at least 15 years before the development of multiscale multiphysics models for steels will be complete.

Open Issues

Although the irradiation modelling papers reflect the reasonably robust state of current understanding, there remain a number of issues to be resolved. First, it is not clear that the assumptions generally involved in current models are sufficiently well supported by mechanical property or microstructural data, for example, the assumption that 'matrix damage' is unaffected by dose rate. This is not entirely certain and will probably remain uncertain until the nature of matrix defects (which may vary with material) is finally resolved. Similarly, the role of phosphorus, and the influence of other factors on phosphorus embrittlement and hardening effects are complex and not entirely clear. In general, the lack of full understanding of irradiation damage mechanisms and the lack of fully relevant experimental data (in particular data for RPV end of life dose rates and irradiation times) will always leave open the possibility for the emergence of unexpected effects. (Over the past 40 years of irradiation embrittlement studies, several "unknown unknowns" have emerged as a result of experimental data – of course even the effect of copper was once an unknown.) In this context more work to directly compare WWER and "western" steels and models may be helpful. The more that models and understanding can be brought together and made universal, the less will be the space for unknown unknowns to exist. In this context, it is not clear that the desire for "simple" irradiation damage models can be fulfilled. The mechanisms of irradiation damage are complex and it may only be through very detailed simulation (multi-scale/multi-physics) models that fully reliable predictions can be produced.

Some issues are not addressed by any of the irradiation damage papers (and indeed are generally ignored or treated as justifiable assumptions by the expert community). The first is the relationship between irradiation damage effects as determined from laboratory tests of mechanical test specimens and the effect of irradiation damage of defects within a vessel. In the latter case, for example, there may be residual stresses, temperature variations, fatigue cycling and other such interactions with the material being irradiated at the crack tip. Second, the papers do not consider the "real world" effects of materials inhomogeneity and uncertainties in the parameters affecting irradiation damage. Improved models may produce more precise predictions, but this precision may be misleading relative to the application – a more accurate understanding of the effect of copper may lead to a less safe situation if the copper in the RPV is inhomogeneous or inaccurately estimated. Thirdly (perhaps a relatively minor point) irradiation effects are usually considered only in terms of changes to the flow properties and grain boundary strength. The effect of irradiation on carbides and other such microstructural features that affect toughness are not considered. Some evidence for carbide effects in irradiation has been reported and might be expected to be important also in annealing/re-irradiation behaviour.

The fracture toughness papers generally focussed on the Prometey model, which is comparable in many respects to the Wallin Master Curve Model (some useful comparisons were presented) but has a more detailed mechanistic basis and can explain apparent changes in curve shape as irradiation hardening increases. One of the papers also provided a method for dealing with the realistic case where the stress intensity factor and materials properties vary along the crack front. This model is more complex than the MC model (but, as the case of the irradiation models, as discussed above, this should be regarded as probably a "fault" of nature rather than something to avoid). Actually, it is not clear whether the model is complex enough: the microstructural changes due to irradiation damage are diverse and complex and loadings on real structures are not generally the same as those on laboratory specimens. Furthermore, the model is empirically calibrated to data and explanation of the wide range of toughness that can be exhibited between different casts of the same material (and within the same case) is outside its scope. These variations are an important consideration in applications, particularly for new build. This perhaps is the direction in which fracture toughness modelling needs to progress. In the case of irradiation damage modelling, the concern is extrapolation of data and the potential emergence of new behaviour that might threaten integrity or undermine safety cases and investment decisions. In the case of toughness, it seems much less likely that the models describing toughness distribution/fracture behaviour as a function of temperature would be undermined in the same way. There are rival models, but they do not give greatly dissimilar results. There is, however, a challenge in understanding the factors responsible for the wide range of toughness values achieved both within and between components. The ability to understand, and control, these variations would potentially enable the production of tougher, more homogenous and consistent components, or at the very least improve the characterization of the toughness of individual components.

Reviewed papers and summaries

- *Grain boundary embrittlement due to reactor pressure vessel annealing, A.V. Nikolaeva, Yu. A. Nikolaev, and A. M. Kryukov, Paper published in the Journal of Nuclear Materials, 211 (1994), 236-243.*

This paper discusses the potentially adverse effect of grain boundary embrittlement during reactor pressure vessel annealing and proposes a model to predict the effect. It starts by discussing the basic mechanisms of annealing of irradiation damage, and pointing out that this may be accompanied by a detrimental effects due to segregation of phosphorus and other elements to grain boundaries (temper embrittlement). The technical challenge is to choose annealing conditions to maximize the recovery of the original properties without inducing segregation. The paper discusses kinetic and thermodynamic models of segregation in some detail and then introduces some experimental data. These were obtained from twelve base or weld Cr-Ni-Mo steels with varying nickel and phosphorus contents. These were heat treated to achieve different prior-austenite grain sizes and then temper embrittled by treating at 420 to 510 °C for 10 to 3000 hours. The degree of embrittlement was measured using Charpy tests. It was found that this was accurately linearly correlated to the model predictions of the grain boundary phosphorus concentration. Also the transition temperature itself could be related to the grain size thus enabling a model for the embrittlement due to segregation taking into account both the degree of phosphorus segregation (dependent on time, temperature, phosphorus and nickel content) and on grain size. The model was found to be consistent with experimental results from materials irradiated in NPPs and then annealed. Full recovery of mechanical properties was not achieved in these cases and this was consistent with model predictions of the temper embrittlement that occurred during the annealing process. The model was therefore proposes as a method for optimizing annealing.

Comments: This is a relatively early paper in this field and remains important and relevant. It describes only a fairly limited number of cases and does not present microstructural evidence to back up the modelling, or discuss the potential role of other chemical elements. This is an active field of work and later papers should also be consulted.

- *About radiation embrittlement kinetics of steels for reactor vessels, P. A. Platonov, E. A. Krasikov, and Ya. I Shtrombakh, Paper presented at the specialist meeting on irradiation embrittlement and mitigation, Espoo, Finland, October 23-26, 1995*

The paper proposes new approaches to the evaluation of radiation embrittlement kinetics, necessitated by the observation that recent data did not follow the accepted Russian standard model for irradiation embrittlement. In particular, low flux RPV operation could produce greater embrittlement than accelerated research reactor irradiations to the same fluence. These effects are explored in a plot of the data versus time, which helps to identify the underlying dependencies and explain why the data that are generally compared with prediction (in the 10 – 15 year operation period) can give good agreement. The discussion is then extended to re-irradiation kinetics following annealing, in particular to predict the early stages of re-embrittlement and the effect of residual shift. These were discussed using mechanistic arguments that may provide an explanation for the unexpected results in the testing of the Novovoronezh NPP Unit 3 Vessel samples. Such mechanistic understanding could also be used to devise annealing schemes in which the post-irradiation embrittlement rate could be controlled.

Comments: this is one of the earliest papers to discuss dose rate effects in the context of mechanisms. Unusually it presents data plotted versus time rather than dose; this is a very effective way of presenting dose rate effects. The understanding of dose rate effects has developed considerably since the paper was published and more recent papers should also be consulted.

- *Radiation embrittlement kinetics of the first generation of WWER-440 RPVs after post-irradiation annealing, P. A. Platonov, Yu. A. Nikolaev, and Ya. I Shtrombakh, Paper published in the International Journal of Pressure Vessels and Piping, 79 (2002) 643-648*

The paper presents a new approach to the estimation of re-irradiation kinetics of WWER-440 RPV steels. Samples cut from several pressure vessels shows much less weld metal embrittlement than predicted by the lateral shift model used in Russian codes. Re-irradiation of such samples, however, gave mixed results, some followed the lateral shift model, and others gave lower shifts. A difference between these irradiation was the ratio between gamma and neutron flux, and previous work by Amaev et el had shown that gamma radiation can reduce radiation damage caused by neutrons, by radiation annealing. A strong gamma ratio effect was illustrated by results from high and low gamma ratio irradiations; the effect was strong because the re-embrittlement was dominated by high phosphorus (the copper effect was low because the annealing had reduced the matrix copper levels to near the solubility limit). A model, incorporating the gamma annealing effect was developed and compared with data. This took into account the effect of gamma irradiation and flux and temperature effects in accelerating the dissolution of cascade-induced phosphorus precipitation. It was found that the model fitted re-irradiation shifts well for 13 out of 14 cases. In the 14th case, the observed re-irradiation shift was much less than predicted. The reason for this was not resolved, but was possibly due to uncertainties in the gamma ratio.

Comments: I do not know whether this uncertainty was resolved, nor whether the gamma radiation mechanism has been validated (I have not read the original Amaev paper). The model description is, however, interesting and worth further study.

- *Advanced method for WWER RPV embrittlement assessment, L. Debarberis, A. Kryukov, D. Erak, Yu. Kevorkyan, D. Zhurko, Paper published in the International Journal of Pressure Vessels and Piping, 81 (2004) 695-701.*

This paper discusses the empirical nature of regulatory (normative) shift models and the mechanisms of irradiation damage in the particular context of WWER-440 steels (see also Debarberis-001 and -002 for similar discussions for the case of RPV steels more generally). A difficult issue for these steels is the role of phosphorus, which appears to vary with copper content. When the latter is high, phosphorus was believed to segregate to the copper-rich clusters, increasing their hardening. When copper is low, phosphorus might form clusters or phosphides. Additional effects included segregation to boundaries or dislocations. The paper then goes on to discuss irradiation re-embrittlement after annealing for these steels. It describes the previous conservative, lateral and vertical shift models, and points out that the data now available show that behaviour can be more complex than had been assumed. In particular, there is a delay to re-embrittlement, which is associated with phosphorus hardening in the absence of a copper effect. This behaviour occurs because annealing puts the phosphorus, but not the copper, back into solid solution. Therefore on re-irradiation there is an effect of phosphorus, but not copper. Both APFIM data and regression analyses of the shift results were consistent with this interpretation. The regression model for re-irradiation gave a reasonable fit to data for different phosphorus contents although there were insufficient data to provide full confidence in the model choice. An important aspect of the work was that it explained the distinctive re-embrittlement kinetics observed at low fluence (a delay to the start of embrittlement) and thus enabled a better interpretation of the data.

Comments: This, and the two following papers, advances the modelling of WWER steels using mechanistic understanding as a basis. In this case, use is made of understanding to explain and predict re-embrittlement kinetics. Work continues in this area, and, as more data and understanding become available, the current model may change. Currently this paper remains both relevant and important.

- *Semi-mechanistic analytical model for radiation embrittlement and re-embrittlement data analysis, L. Debarberis, A. Kryukov, F. Gillemot, B. Acosta, and F. Sevini, Paper published in the International Journal of Pressure Vessels and Piping, 82 (2005) 195-200*

This paper presents a practical model for fitting and interpreting irradiation data. It has a sounder mechanistic basis than the very simple empirical models used for some regulatory models, but is easier to use than some of the more complex analytical models developed in recent years. The model assumes (following conventional practice) that there are three independent components of irradiation damage: matrix damage; precipitation damage and segregation (non-hardening embrittlement). Each of these components is modelled using a simple analytical expression, with a total of six adjustable parameters. The advantages of the model were that it can better fit certain data sets (because it more accurately represents the underlying mix of mechanisms) and that it can explain (in conjunction with microstructural investigations) observed re-irradiation kinetics. The model was tested using a range of data and was found to reproduce the observed trends well, even with simple assumptions about the dependency of the respective parameters on the copper and phosphorus contents.

Comments: the modelling of irradiation shift in WWER steels is treated here along much the same lines as those adopted by Eason Wright and Odette and others for Mn-Mo-Ni steels. This is, I believe, the correct way to go and this is an important paper in advocating this approach and demonstrating its success. In the case of Debarberis et al, compared to Eason et al, the models used are rather simpler. Although this is desirable from a practical point of view and entirely appropriate when the database is relatively limited, it may be that, as more data are obtained, more complexity will be required. It would be interesting to see a comparison between the two models.

- *Fluence rate effects on irradiation embrittlement of model alloys, L. Debarberis, F. Sevini, B. Acosta, A. Kryukov and D. Erak, Paper published in the International Journal of Pressure Vessels and Piping, 82 (2005) 373-378*

This paper provides a brief summary of the mechanisms of irradiation damage and identifies the importance of the development of understanding to plant lifetime assessment. It also summarizes the model alloys study at JRC-IE in which seven model alloys with varying levels of nickel, copper and phosphorus were irradiated in the LYRA facility in the High Flux Reactor at Petten. The paper then describes an extension to this work in which seven of the alloys were irradiated in the Kola NPP (Russian Federation) and five were irradiated in the Rovno NPP (Ukraine). These were low flux surveillance channel irradiations and hence enabled the effect of neutron flux (fluence rate) to be assessed. Interpretation of the data was done through a semi-mechanistic irradiation shift model (see Debarberis-002). This confirmed that flux effects are strongest for lower fluences and for high copper, nickel and phosphorus.

Comments: It is important that fluence rate (flux) effects are properly taken into account in analysing data and prediction future trends and this paper shows this for the case of WWER steels. Usefully, it identified the range of conditions over which flux effects may be expected. It does, however, remain to be seen whether the models are sufficiently detailed, and the assumptions (for example the invariance of matrix damage with flux, which has recently been challenged, justified).

- *Thermally activated deformation of irradiated reactor pressure vessel steels, J. Böhmert and G. Müller, Letter to the Editors of the Journal of Nuclear Materials, 301 (2002) 227-232*

The letter discusses experimental data and analysis to determine the parameters for thermally activated deformation of two irradiated and unirradiated WWER 1000 steels with somewhat different nickel contents. It starts by describing how the parameters involved in the Arrhenius rate equation can be determined from tensile tests at different (varying in steps) strain rates and temperatures. Next, it describes the experimental details, and finally the results. These show that the activation volume and activation enthalpy are strongly temperature dependent (as expected and assumed). However, there was no effect of irradiation or of difference between the two welds. This shows that irradiation produces long-range obstacles, hence an athermal hardening effect. It was therefore concluded that the concentration of interstitials is not changed by irradiation and that irradiation does not change the shape of the yield stress versus temperature curve. It is also concluded that the difference in nickel between the welds (1.11% and 1.59%) does not affect any thermal obstacles.

Comment: This is a useful paper providing a rationale for considering the effect of irradiation to be only on the athermal hardening component. This is an active area of current research and discussion therefore later papers should also be consulted. This paper discusses only nickel, but other elements, such as Si and Mn should perhaps also be considered.

- *Radiation embrittlement modelling for reactor pressure steels: I. Brittle fracture toughness prediction, B. Z. Margolin, V. A. Shvetsova, and A. G. Gulenko, Paper published in the International Journal of Pressure Vessels and Piping, 76 (1999) 715-729*

This paper starts by discussing three principal methods of determining irradiated fracture toughness predictions: the Charpy shift, Master Curve and the local fracture criteria models. The paper then goes on to describe in some detail the last of these (the Margolin et al/Prometey) models for both brittle failure (their model for ductile failure is described in Part II of the paper (not included in the modelling folder). The model requires seven parameters and methods for obtaining these from unnotched and notched cylindrical specimens are discussed; values are given for 2.5Cr-Mo-V steels. The effects of irradiation are then considered: first on the athermal component on yield stress and on strain hardening. The latter is shown to be insensitive to irradiation. Second the effects of irradiation on the parameters affecting brittle fracture are considered. Although the critical brittle fracture stress may not be affected by irradiation, unless intergranular fracture is promoted by temper embrittlement during annealing, the initiation of cracks from carbides may be affected by phosphorus segregation to the carbide/matrix interfaces. Predictions using the model show significant changes to the shape of the K_{IC} versus temperature curve with increasing irradiation and this limits the range over which the assumption that irradiation simply shifts the curve laterally is true. The model also predicts a non-additive (synergistic) effect of phosphorus and copper, increasing scatter for irradiated steels, and a larger size effect for irradiated steels. Finally comparisons are made between the Margolin et al and the Richie Rice and Knott and Beremin models. It is shown that the fracture criteria used in the last two of these do not adequately account for the data.

Comments: the development of the Prometey model has been a very important "challenge" to the Master Curve model, which has almost become the orthodoxy for US code based RPV integrity assurance. The Prometey model is more complex and has arguably a more detailed mechanistic rationale and this enables it to describe a wider range of behaviour. The problem for modelling in this area is that very large amounts of high quality data are required to validate models empirically, and it is very difficult to validate them mechanistically. This is a very active area of work around the world, and although this paper is not outdated, recent work should also be consulted.

- *New approaches for evaluation of brittle strength of reactor pressure vessels, Boris Margolin, Eugene Rivkin, George Karzov, Victor Kostylev and Alexander Gulenko, Paper published in the transactions of the 17th International Conference on Structural Materials in Reactor Technology (SMiRT 17), Prague, Czech Republic, August 17-22 2003, Paper # G01-4*

This paper discusses, in the context of the Master Curve approach, the practical issues of how shallow crack, biaxiality and crack front length effects should be treated in fracture integrity assessments. The discussion of crack front effects considers the case where the SIF and the K_{IC} vary along the crack front (for example for a semi-elliptic surface breaking defect, and also the more complex situation where the loading is non-monotonic, non-isothermal. For the latter, parts of the crack front may experience pre-loading, analogous to warm pre-stressing, whereby for part of the transient failure will not occur. Use of these approaches can improve the accuracy of structural integrity assessments and reduce conservatism.

Comment: this paper deals with the very practical issue of the integrity assessment of "real cracks".

- *Fracture toughness prediction for RPV steels with various degrees of Embrittlement, Boris Margolin, Alexander Gulenko and Victoria Shvetsova, Paper published in the transactions of the 17th International Conference on Structural Materials in Reactor Technology (SMiRT 17), Prague, Czech Republic, August 17-22 2003. Paper # G03-4*

This paper compares the Master Curve and Prometey (Margolin et al) model approaches to fracture toughness prediction. In particular it examines the invariance of the toughness versus temperature curve shape in the Master Curve approach versus the varying shape in the Prometey model. For unirradiated material and for low irradiation shifts, the two models give similar results. However, it was shown that, for a WWER base metal highly embrittled by a special heat treatment to give a ΔT41J of 180 °C, the Prometey model provided a better description of the data. The flattening of the Prometey model curves for highly embrittled material was shown to be consistent with local fracture models. The paper concludes by discussing the shape parameter (b) in the two parameter Weibull distribution. It is shown that departures from b=4 can be justified on theoretical grounds and supported by data. In highly embrittled material, much higher values may be applicable. However, the authors emphasise that, for the three parameter Weibull model, b=4 works well in both the transition range and lower shelf.

Comments: A useful aspect of this paper is that it does show comparisons with the Master Curve and identify conditions in which the two approaches diverge. This is helpful to identify why one should be preferred rather than the other, and is a useful change from papers that do not explicitly consider other work.

6 Surveillance (Surveillance)

Session Summary

The collection of papers dealing with the data of reactor pressure vessel (RPV) materials degradation due to the irradiation have been reviewed with the aim to follow the evolution of this interesting field of research in the last decade. This approach enables:

- to recognize the leading research organizations in the field,
- to have a clear picture of the advances achieved so far,
- to determine the existing open issues,
- to identify future research activities needed.

The collection of papers reviewed is focused on the SSP testing results of reactor pressure vessel steels typical for the WWER technology. The main research activities described in the papers reviewed are focused on the following aspects:

- Development of new modernized type of surveillance specimen programs.
- Validation of the new evaluation methodology for WWER RPV's material.
- The irradiation environment monitoring improvement.

- Comparison of transition temperature shifts fracture toughness due to the irradiation from the surveillance specimen programs of several laboratories mostly obtained from standard Charpy specimens testing.
- Application of embrittlement trend curves based on fracture toughness data, mainly using the master curve approach.

In most of the 17 papers reviewed the results were obtained from the WWER-440 technology. The results were evaluated in several research organizations (VUJE, VTT, RRI Kurchatov, NRI Rez, SKODA). Specific summaries of the papers reviewed are given below.

Consolidated Conclusions

- For both technologies WWER-440 and WWER-1000 modernized types of surveillance programs were developed, which are used in operated units. According to the planned and step by step realized power up rate of older units and new fuel generation utilization are under preparation specific SSP in Russia, Hungary, Finland, Czech Republic and Slovak Republic, with aim to monitor these new operational conditions.
- In the last ten years new methods for evaluation of irradiation influenced WWER RPV`s material mechanical properties changes were validated or are under validation.
- As important part of SSP mentioned above is continuous effort for improvement of neutron fluence and irradiation temperature measurement precision.
- Due to the fact that most data for the transition temperature shift are (were) obtained from standard Charpy specimen tests, it would be useful to find the proper correlations with the sub-Charpy testing results.
- In current SSP is accented the effort to use as the main method for the brittle fracture temperature shift evaluation the fracture toughness testing with the Master Curve approach application.

Open Issues

Against the fact, that great effort was put into surveillance programs improvement several issues are still not clarified:

- how to estimate the plants operation extension above the original lifetime when there are not, or is a great lack of experimental materials,
- the procedures for evaluation of power uprate influence,
- how to take into account the difference in neutron flux for RPV and surveillance
- the procedures for evaluation of new fuel generation implementation influence,
- what are the properties of high irradiated heat affected zone materials around the RPV's welds and below the austenitic cladding,
- how to test the irradiation embrittlement of reactor internals.

To analyze and find the answers for these issues is great challenge for the future research projects and for the new surveillance programs too.

Reviewed papers and summaries

- *Surveillance programs and irradiation embrittlement research of the Loviisa nuclear power plant - Ahlstrand R., Törrönen K., Valo M., Baers B., Surveillance Embrittlement of Nuclear Pressure Vessel Steels: An International Review (Second Volume), ASTM STP 909 (1986) 55-69*

The detail description of the Loviisa units 1 and 2 surveillance specimen programs is the topics of this publication. The details of specimen types, irradiation schedules and analysis from the standard program results are presented too. Influence of impurities in the weld metal samples was studies by using several empiric equations. The effect of the core modification to the transition temperature shift is involved too. With aim to improve the results obtained form the standard program were prepared:

- new sets of specimen for irradiation,
- new CT-specimen of round type,
- several new type of neutron monitors new type of fluence calculation and evaluation.

The influence of thermal annealing of Loviisa unit 1 was studied too, by micro hardness measurements.

- *Assessment of irradiation conditions in WWER-440(213) RPV surveillance position - Ballesteros A.,Bros J., Debarberis L., Sevini F., Erak D., Gerashchenko S., Kryukov A., Shtrombakh Y., Goloschapov S., Ionov, Pytkin Y., Anikeev Y., Banyuk G., Plush A., Gillemot F., Petrosyan V., ICONE 12, 12th Int. Conf.on Nuclear Engng., April 25-29,2004, USA, paper 49477*

- *Consolidation of scientific and technological expertise to assess the reliability of reactor pressure vessel embrittlement prediction in particular for the arctic area plant (COBRA) - Ballesteros A.,Bros J., Debarberis L., Sevini F., Erak D., Gezashchenko S., Kryukov A., Shtrombakh Y., Goloschapov S., Ionov, Pytkin Y., Anikeev Y., Banyuk G., Plush A., Gillemot F., Tatar L., Petrosyan V., Nuclear Engineering and Design 235(2005)411-419*

The evaluation and prognosis of reactor pressure vessel (RPV) material embrittlement in WWER`s and the allowable period of their safe operation are performed on the basis of impact test results of irradiated surveillance specimens. The main problem concerns the irradiation conditions (irradiation temperature, neutron flux and neutron spectrum) of the surveillance specimens that have not been determined yet with the necessary accuracy. The COBRA project was designed to solve these problems. Surveillance capsules were equipped with state of art dosimeters and temperature monitors (melting alloys). Thermocouples were installed to measure directly the irradiation temperature in the surveillance position during reactor operation. The selected reactor for the experiment was the Unit 3 of Kola NPP situated in the arctic area of Russia. Irradiation of capsules and online temperature measurements were performed during one fuel cycle. The temperature of irradiated surveillance specimens in WWER-440/213 reactor can be accepted as 269.5 ± 4°C. Maximum neutron flux in the Charpy specimen simulators, equals ~2.7 x 10^{12} cm^{-2} s^{-1} with E> 0.5 MeV. It was established that depending on the orientation of the capsules with respect to the core, the detectors of the standard surveillance capsules could give both overestimated and underestimated neutron flux values, which can reach 10%. It is necessary to carry out gamma spectrometer examination of each specimen, with subsequent sampling of the specimen to determine the 54Mn activity, in order to obtain correct and accurate data of the neutron flux received by the surveillance specimens. The project results are not only of interest for the Unit 3 of Kola NPP but also of application to all the operating WWER-440/213 reactors.

- *Surveillance Specimen Programmes for WWER reactor vessels in the Czech Republic - Brumovsky M.,Erben O., Kytka M., Novosad P., Int.Symposium FONTEVRAUD 5, 23-27 Sept.2002*

The main parameters of continuous monitoring of RPV's degradation for the whole lifetime are one of important elements of safety and reliability of nuclear units. Paper describes a present state of material degradation in reactor pressure vessels of WWER type reactors manufactured in Czech Republic. Standard surveillance program for WWER-440 V-213 type reactors is described and discussed its deficiencies together with main results obtained. New supplementary surveillance program was developed and started with quite new design for WWER-1000/V.320 type reactor pressure vessels. Description of material selection, containers design and location as well as withdrawal plan connected with ex-vessel fluence monitoring is described too.

Together with the description of Standard Survei1lance Programs and their disadvantages are also shown the Supplementary Surveillance Program for NPP Dukovany with WWER-440 type and Modified Surveillance Program for NPP Temelin with WWER - 1000 type reactors.

A proposal is given to use the NPP Temelin reactor as a "host" reactor for an Integrated Surveillance Program for WWER- 1000 reactor pressure vessels.

- *Irradiation-induced structural changes in surveillance material of WWER-440-type weld metal - Grosse M., Denner V., Boehmert J., Mathon M.-H., Journal of Nuclear Materials, 277(2000)280-287*

The application of small angle neutron scattering (SANS) and anomalous small angle X-ray scattering methods (SAXS) was performed for the investigation of the irradiation-induced micro structural changes in surveillance materials of the WWER 440-type weld metal Sv10KhMFT. Due to the high neutron fluence, a strong effect was found in the SANS experiment. No significant effect of the irradiation is detected by SAXS. An analysis of the SAXS shows, that the spectra scattering intensity is mainly caused by vanadium-containing (VC) precipitates at grain boundaries. Neutron irradiation produces additional scattering defects of a few nanometers in size. Assuming these defects are clusters containing copper and other foreign atoms with a composition according to results of atom pro be field ion microscopy (APFIM) investigations, both the high SANS and the low SAXS effect can be explained. All results of investigations were compared, three types of scattering inhomogenities are presumed:

- vanadium carbides VxCy,
- grain boundaries,
- clusters containing copper, manganese, silicon, nickel and phosphorus.

Grain boundary scattering and the scattering at the vanadium carbides were observed. In the irradiated state an additional scattering contribution can be observed by SANS but not by SAXS. Assuming the irradiation-induced scattering is caused by clusters of a type that was identified by APFIM as an irradiation defect, all measured contrast variation effects can be explained. These results should be verified by thermodynamic investigations using the annealing behaviour of the irradiation defects. APFIM investigation at the same material would be helpful to confirm the results. Full understanding of all irradiation-induced damage in the reactor steel is still not achieved.

- *Problems in a standard surveillance programme of WWER-1000 reactor pressure vessels - Brumovsky M.,Rieg C-Y., IAEA meeting on Irradiation Embrittlement and Mitigarion, Espoo, Finland, October 23-26,1995, IWG-LMNPP-95/5*
- *Surveillance Specimen Programmes for WWER reactor vessels in the Czech Republic - Brynda J.,Hogel J., Brumovsky M., Trans.17th Int.Conf.on Structural Mechanics in Reactor Technology (SMIRT 17), Prague, August 17-22(2003), paper G02-5*

The deficiencies of the standard WWER-1000 surveillance program and activities necessary for their overcoming are discussed in both papers. The general problems related to the accuracy of RPV lifetime assessment are defined on the base of the detailed analysis of standard surveillance program results and the experience acquired during WWER-1000 reactors exploitation. It is shown that the main surveillance program deficiencies are related to the containers design, the inhomogeneous neutron field on surveillance specimens in one testing set, the noncorrectness of the applied method for irradiation temperature measurement and the low value of lead factor for first assembly row. Several conceptual programs are developed and presented with aim to clarify and solve Standard Surveillance Program problems of operating and still non-operating WWER 1000 plants using experience and contemporary knowledge of irradiation embrittlement process.

The actions foreseen in the programs are related to:

- improvement of the irradiation temperature measurement by implementation of new type monitors of low melting eutectic alloys;
- more precise characterization of azimuthal and axial neutron field by application of sets of neutron monitors with expanded specification and different shape, gamma scanning of all specimens and neutron flux measurement in the RPV cavity. The homogenous neutron flux on specimens is assured by a new design of the irradiation container.

Extension of Surveillance Database is proposed by implementation of additional surveillance programs. It foresees irradiation in the specific reactor, in a test reactor or in a host power reactor of tensile, Charpy CV and pre-cracked Charpy specimens manufactured from BM, WM, HAZ and RPV cladding and reference materials. The application of reconstitution technology is proposed for specimens also.

- *Surveillance programme for WWER-440/type 213 reactor pressure vessels - Standard programme, re-evaluation of results, supplementary programme -Brumovsky M.,Novosad P., Zdarek J. , Effects of Radiation on Materials, 17th Int.Symposium, ASTM STP 1270,1996, 522*

Detailed analysis of the Standard surveillance program applied in Czech Republic for monitoring of neutron-induced embrittlement of WWER 440/213 RPV's is the topic of paper. Activities necessary for elimination of Standard program disadvantages are discussed and new measures are planned for their realization.

Re-evaluation of neutron fluence by measurement of specimens individual activity and increasing the number of specimens with similar fluence in one test set by reconstitution of Charpy specimens is proposed in order to improve the precision of fracture mechanic calculations of RPV resistance against brittle fracture.

The design of two new chains for experimental verification of flux gradient on outer chain capsules and irradiation temperature measurement with accuracy required by the standards are discussed. The method for reconstitution of static and dynamic fracture toughness test specimens is verified by these experiments also.

A WWER-440 Supplementary surveillance program based on the experience gathered during standard surveillance program fulfillment, on the verification experiments results and on the contemporary requirements for reliable assessment of RPV residual lifetime. The program provides irradiation of inserts from WM and BM for reconstitution of Charpy impact and static fracture toughness tests specimens as well as of inserts from the inner and outer RPV cladding layers. Two sets of neutron monitors with extended specification installed in Al and Gd tubes are provided for more precise determination of fluence on surveillance specimens. Additional monitors with different shape are applied for axial and azimuthal fluence distribution determination. The accuracy of irradiation temperature measurement is improved by implementation of low melting alloys monitors.

The new program supplies additional important data necessary for RPV integrity assessment and assures monitoring of primary embrittlement during the whole planned RPV rest lifetime. Investigation of re-embrittlement rate is provided in the program too.

- *Irradiation embrittlement monitoring of WWER-440/213 type RPV's - Kupca L., Beno P., Nucl. Engineering and Design 196(2000)81-91*
- *Irradiation embrittlement monitoring programmes in the Slovak Republic nuclear power plants - Kupca L., Int. Symposium FONTEVRAUD 5, 23-27 Sept.2002,*

The activities of four Surveillance programs developed for the RPV metal embrittlement monitoring of the WWER-440/213 energy units in Slovak Republic are discussed in the papers. Some results for the changes of BM and WM mechanical properties from the Standard surveillance program (SSSP) are presented also.

The high "lead factor" of the irradiated specimens, the incorrect irradiation temperature determination, the low precision of n-flux measurement and fluence calculation, the inaccurate determination of transition temperature, the short period of program duration are defined as main limitations of SSSP. An extended Surveillance Program (ESSP) designed by VUJE Institute was applied in Jaslovské Bohunice NPP V-2. New Modern Surveillance program (MSSP) for Mochovce Unit 1 and Unit 2 RPV`s where the monitoring of RPV's irradiation embrittlement is provided till the end of Unit designed lifetime is presented in the papers.

Irradiation of inserts from BM and WM for manufacturing of reconstituted Charpy-V and COD test specimens is foreseen in ESSP and in MSSP. Small punch testing and sub-size Charpy specimens are proposed for irradiation in MSSP. The neutron fluence determination on surveillance specimens is improved by extension of neutron monitors specification, increasing of monitors' number and by additional RPV outside dosimetry. More precise irradiation temperature measurement is achieved by implementation of temperature monitors from low melting alloys and of thermocouples. The chains are installed in RPV in position with lead factor limited to 3-5. The new design of irradiation chains, the more precise fluence determination and the application of specimen's reconstitution technology results in improved accuracy of transition temperature determination.

Investigation of irradiated specimens from BM, WM and HAZ metal by transmission electron microscopy, Mössbauer spectroscopy and electron-positron annihilation is foreseen in the programs. NRA SE approves both programs and their fulfillment has started.

- *Extended analysis of WWER-1000 surveillance data - Kryukov A., Erak D., Debarberis L., Sevini F., Acosta B., Int.J.Pressure Vessels and Piping 79(2002)661-664*

The main task of the paper is to derive an empirical dependence of critical temperature shift on neutron fluence, considering Ni an Mn concentration in 10ChGNMAA RPV weld metal with low concentration of Cu and P. For this purpose a correlation analysis of existing database with results from WWER1000 surveillance specimens is performed. The specimens are classified into three subgroups: High Ni-High Mn; High Ni–Medium Mn; Low Ni.

The authors find that all welds of subgroup "Low Ni" show much lower embrittlement rate than predicted by Russian Guide. The guide prognosis remains conservative for most of the welds of "High Ni – Medium Mn" subgroup also. The embrittlement rate of subgroup "High Ni - High Mn" is much higher than expected by Russian Guide. The determined threshold for nickel effect is Ni>1.5 wt% and for Mn effect - Mn > 0.8 wt%.

The specific ΔTkf data sets are analyzed by statistical correlation method using equation ΔTkf = (aNi + bMn + c). F1/3 and the ΔTkf predictive capability within 20oC scatter band are established. It has to be mentioned that no values of the correlation parameters a, b and c are given in the paper. Recommendations for future activities in order to reduce the uncertainty of determination are offered in the paper conclusions.

- *Surveillance of WWER-440C reactor pressure vessels - Brumovsky M., Pav T., Radiation Embrittlement of Nuclear Reactor Pressure vessel Steels: An International Review(Fourth Volume), ASTM STP 1170, 1993, 57-70*

The documents describes standard surveillance programme (SSP), which was prepared, according to the Russian project documentation. Specimens for the surveillance program were produced of the top part of the ring and of the weld that is placed in the RPV area opposite the active zone, the withdrawal was located in the places with minimal ¼ of the individual RPV wall. For the SSP were used tensile, Charpy-V and COD specimen from the base metal, weld metal and from heat affected zone. The placement of the individual irradiation capsules is described, including the neutron fluence measurement. The number of capsules in individual chains was 19 or 20 in the AZ area and 6 or 7 in the place above the reactor core. Withdrawal schedule of the capsules and following evaluation was in this programme after 1, 2, 3, 5 and 10 years exposure. The evaluation of irradiated samples was performed in NRI Rez hot cells. The criteria for evaluation of the transition temperatures, including the shift of the critical temperature were determined according to the Russian standards.

First results from the surveillance programmes of the 6 reactor blocks WWER 440 type V-213 operated in Dukovany and Bohunice NPP were used for the evaluation of the trend curves by calculation from the chemical composition (content of P and Cu) according to NRC Reg. Guide 1.99, Rev.1, with conclusion that the results are in all cases comparable.

It was found that the shift of the transition temperature is better described with the exponent ½ (according to the NRC Regulatory Guide 1.99, Rev.1), than of exponent 1/3 used in Russian standard formula Δ F = AF x (F x 10-22)1/3 . The rates of the transition temperature shift Δ F of the material RPV are under the trend curves designed for this evaluation with Russian standard.

- *Comparison of static and dynamic transition temperature shifts in WWER reactor pressure vessel steels - Brumovsky M.,Falcnik M., Kytka M., Malek J., Novosad P., Effects of Radiation on Materials, 19th Int.Symposium, ASTM STP 1366, 2000, 56*

The evaluation of reactor pressure vessel material properties based on fracture mechanics approach with use of static (KIC) and dynamic (KId, KIA) fracture toughness trends is topic of this paper. The comparison of both criteria embrittlement assessment used in WWER Code (PNAE G-7-002-86) and ASME Code Section III, Div.1 is presented too, for the results from standard surveillance programme of WWER-440/V-213C reactor pressure vessel materials after five (base material) and ten (weld metal) years of NPP operation. Second part of the programme was concentrated on the steel 15Kh2NMFA for WWER 1000/V-320 type reactor pressure vessels focused on the effect of different contents of copper and phosphorus in the steel. For this experiment was prepared six laboratory heats this type of steel with different content of phosphorus (two level – 0,012 and 0,021%) and copper (three level – 0,08, 0,30 and 0,50%).

Irradiation was performed in the experimental reactor LVR-15 at temperature 288°C to a neutron fluence equal to 2,6 x 1024 n.m-2 (neutrons with energies above 0,5 MeV).

The reconstitution technique for broken halves of standard surveillance Charpy specimens with using electron beam welding method is presented. This technique was proven as a very effective and valuable method for re-evaluation of specimens from the standard surveillance programs. Very important is using of the same specimens, and certainly fully identical conditions for the different tests. Nevertheless, difference still exists between static transition shifts from reconstituted specimens and from direct tests of pre-cracked Charpy specimens within the original surveillance programme.

All results from both RPV WWER materials 15Kh2MFA and 15Kh2NMFA are summarized in this paper. Results from base as well as from weld metals showed that a linear dependence between shifts of static fracture toughness and Charpy notch impact exists. There is practically no difference between transition shifts from different dynamic type tests, i.e. shifts from Charpy impact and dynamic fracture toughness tests. Experiences obtained within these tests were used for Supplementary Surveillance Programme design in Dukovany NPP.

- *Supplementary surveillance programme for reactor pressure vessels of WWER-440/V-213C type reactors in NPP Dukovany - Brumovsky M.,Erben O., Novosad P, Effects of Radiation on Materials, 19th Int.Symposium, ASTM STP 1366, 2000, 220,*

- *Supplementary surveillance programme for WWER-440/V-213C reactors - Brumovsky M., Erben O., Novosad P., Proceedings of ICONE 5; 5th Int. Conference on Nuclear Engineering, May 26-30, 1997, Nice, 1*

Re-evaluation of standard surveillance specimen programs for NPP Dukovany and design of a new Supplementary Surveillance Programme are described in this report. The critical analysis of a standard surveillance programme for WWER-440/213 type reactor pressure vessels is focused on facts that:

- surveillance specimens were located in high lead factor positions (approx. 12 for base material and 18 for weld metal) and thus only up to about five years are suitable for surveillance,
- specimens from heat-affected zone do not fully represent material from the active core region as welding coupons are made from special manufactured plates,
- there are no tensile specimens for fracture toughness test,
- capsules were designed only for two Charpy impact type specimens, thus at least 6 capsules are necessary for testing of one set of specimens (min.12 pieces for transition curve,
- diamond irradiation temperature monitors were shown to be practically unusable, as changes in its lattice parameter depend also on neutron flux, not only on fluence. This effect also influences its behavior during annealing,
- correction factors may have to be applied but their uncertainties are quite still large.

The second part of this paper is focused on the base principles and proposals of new surveillance programme for WWER-440/V-213C reactor pressure vessel materials in Dukovany NPP.

Design of this Supplementary Surveillance Programme enables monitoring of neutron fluences and changes in material properties during the whole remaining reactor pressure vessel lifetime. Irradiation with a low lead factor (in the range between 2 and 3) allows applying the results from this surveillance programme for the RPV embrittlement assessment. Each capsule contains twelve inserts (10 x 10 x 14 mm) for further reconstitution of one set of specimens to cover the whole transition curve (Charpy-V notch toughness or static fracture toughness type specimens). All capsules contain melting temperature monitors base on low-melting alloy of Pb, according with ASTM E 1214-87 and KTA 3203-84 codes and sets of neutron activation (Fe, Cu, Nb) and fission (U, Np) monitors. These monitors in a form of a foil and wires are located in full axial as well as azimuthal directions inside a capsule. These measurements are extended by gamma scanning of all specimens after their withdrawal.

Materials for RPV specimens are only of archive type, i.e. the same as for standard surveillance programme. Moreover, IAEA reference material (JRQ type SA 533-B from IAEA programme) is used for comparison of results in different reactors and materials. Also cladding materials is insert to irradiated capsules, because its behavior is important for RPV integrity assessment during pressure thermal shock (PTS) regimes. A part of specimens is determined for annealing efficiency and re-embrittlement rate of real reactor pressure vessel material after recovery annealing.

During each reactor campaign is performed ex-vessel neutron measurements in cavity with aim to compare results from surveillance as well as from ex-vessel position every year of operation and supplementary calculations. This supplementary surveillance programme was started in 1997 by loading the first chains with capsules on Dukovany NPP.

- *The properties of WWER-1000 type materials obtained on the basis of a surveillance program - Kryukov A.M., Nikolaev Yu., Nucl. Engineering. And Design 195(2000)143-148*

This paper was published in Nuclear Engineering and Design journal in the year 2000 and describes the surveillance tests results of the reactor pressure vessels of three Russian WWER-1000 units. All these data were obtained in the original surveillance programme conditions where the surveillance capsules are located in a position with a high neutron flux gradient, moreover, the disadvantages of the surveillance capsule design (location of the specimens in positions not in contact with each other) and the difference between maximal and minimal flux values were 160–170%. As well the irradiation temperature of the surveillance specimens is 300–310°C, i.e. 10–20°C higher than the irradiation temperature of the RPV walls. Surveillance specimen materials were manufactured from steel 15Kh2NMFAA and welds from weld metal Sv-10KhGNMAA. Nevertheless the materials provided a good basis to study the effects of nickel contents since it was possible to analyze the test results measured with steels having various nickel contents and also some welds had very high contents of nickel (1,21 to 1,88 wt.%). The tendency of RPV materials to brittle fracture was estimated with the help of the ductile-to-brittle transition temperature (DBTT) obtained from the impact tests of Charpy-V specimens. The DBTT of materials in both as-received and irradiated conditions were evaluated by a hyperbolic tangent function.

There are discussed trend curve dependence of the DBTT shift on fast neutron fluence and effect of nickel on radiation stability especially of weld metal. The embrittlement rates were compared to those predicted by the Russian Regulatory Guide. This trend curve strongly underestimates the embrittlement of some materials. The correlation between the observed and predicted data seems to be low, too.

The inclusion of any additional metallurgical variables in the trend curves was not found to significantly affect the scatter of the experimental data around the values predicted. Significant effect of nickel content on irradiation embrittlement sensitivity was found for the weld materials. The nickel effect was not observed for the WWER-1000 RPV base materials and heat affected zone of the welds. The fact that the trend curves of the base metal and HAZ did not depend on the chemical composition can be related to the low variation of the nickel content in these materials and the relatively low nickel content in comparison with the weld metals. The small variations of the residual impurities (phosphorus, copper, etc.) in the investigated surveillance materials disable to determine any dependence of the radiation response on the residual impurities contents. In conclusion of this paper stated that prediction for the irradiation embrittlement to the end-of-life fluence may not be reliable for the high nickel materials.

- *Tests of surveillance specimen containers for WWER-1000 reactor pressure vessel on cyclic strength with the aim to assure their operation lifetime - Bykov E.M., Popadchuk V.S., Zhukov R.Yu., Ponomarev A.M., 3rd NTK on "Assurance of NPPs with WWER", Podolsk, Russia, 26-30 May 2003, Vol.2, 96*

Modernized surveillance specimens programme for reactor pressure vessels WWER-1000 used flat type irradiation containers. The containers are made of austenitic stainless steel 08CH18N10T type sheets welded together in a box. Thickness of the sheets is two millimeters. The surveillance specimens for tensile test, Charpy-V impact test, pre-cracked impact test and CT-0,5 for fracture toughness assessment are used in this program.

The paper describes special cyclic tests of these containers for its design lifetime verification performed in the special experimental loop with primary circuit environment in autoclave with volume 0,07 m^3.

In the next part of the paper is presented the analysis of obtained results and their evaluation. For investigation were used following methods:

- visual test (magnification till 50)
- leakage test (helium)
- dimensions verification (micrometer)
- penetration test of the welds and heat affected zones
- metallographic analysis (light microscope with magnification 200 times)
- micro hardness measurements.

Results of all these tests are summarized to the following conclusions:

- No defects and any other damage were found neither for base material of the containers nor in welds and thus operational lifetime of forty years is assured.
- Dimensions change of the irradiation container is negligible and do not exceed the measured value: 0,032 +-0,001 mm.
- The tests confirmed the operation reliability of the welded flat type containers made from austenitic stainless steel, which will be used for surveillance programme of reactor pressure vessel materials WWER-1000 type reactor.

7 Fracture Toughness (Property-Property Correlation)

Session Summary

The collection of papers reviewed is mainly focussed on fracture toughness testing of reactor pressure vessel steels typical of WWER reactor type. In 5 of the 8 papers reviewed the Master Curve methodology is applied. The papers cover the period 1999-2005. During this period there is a consolidation of the Master Curve approach for Western PWR RPV materials and many research organizations (VTT, the Ukrainian Nuclear Research Institute, NRI Rez, etc.) were trying to validate this new approach using Russian RPV materials. In several papers (Ballesteros, Brumovsky, Valo, Viehrig) the authors tried to develop correlations between the standard Charpy reference temperature and the master curve temperature T0. These correlations are preliminary due to the limited number of data. The results included in this set of papers are important contribution to the WWER material database on Fracture Toughness. Nevertheless, as several authors recognize, more research and data are needed to establish reliable embrittlement trend curves, which can be used for accurate determination of the RPV lifetime, facing in particular long term operation. Specific summaries of the papers reviewed are given below.

The research activities described in the papers reviewed are focussed on the following aspects:

- Development of reference fracture toughness curves.
- Validation of the master curve methodology for WWER RPV material.
- Comparison of shifts in fracture toughness due to irradiation with those obtained using standard Charpy specimens.
- Development of embrittlement trend curves based on fracture toughness data, mainly using the master curve approach.

Consolidated Conclusions

The Ames Network has made some conclusions on the Through-life toughness aspects, which are shared by the participants of the workshop:

- The accuracy of through-life toughness bounds can be greatly increased leading to higher toughness predictions.
- But can we be sure that no important factors have been overlooked?
  - Assumptions in TLT prediction:
    - Invariance of curve shape
    - Independence of SOL and Δ
    - Independence of uncertainties
  - Accuracy and completeness of uncertainty estimation
  - Toughness in other regions of the vessel
  - Scale and stress-state factors
  - Transference from experiment to reactor
    - Real life defects
    - Crack tip stress state and environment
    - Material homogeneity

There are two different views on the Master Curve (from Russia and from the rest of the world).

It is recognized that the number of papers reviewed in this demonstration exercise is very limited to assert any categorical conclusion, but some further references taken into consideration point out conclusions in the same direction.

Open Issues

Given the nature and contain of the papers reviewed, it is clear that macroscopic fracture behaviour needs more fundamental underpinning. For example, the following questions need to be answered:

- What defines the master curve?
- Does the master curve fit into a larger frame?
- How to effectively treat lower boundaries?
- How to define and use fracture toughness trend curves?

The papers reviewed confirm that most of the start-of-life issues are known unknowns (i.e., we know enough), but there is scope for the existence of unknown unknowns. For instance, deviations from the master curve behaviour can not be ruled out until a fully physical basis for it has been established.

Further, there is no trend curve of the Master Curve, there is not enough data existing and fracture toughness testing is not the normal way for surveillance.

Reviewed papers and summaries

- *Applications of the Master Curve Approach to Irradiated Steels, A. Ballesteros, J. Bros and M. Brumovsky, paper presented at the 12th International Conference on Nuclear Engineering, Arlington, Virginia, USA, April 2004.*

This paper discusses the current situation (in April 2004) of the Master Curve methodology. The open issues are discussed and results from the Spanish project CUPRIVA and Tacis project IRLA are presented. The application of the master curve methodology to the Spanish reactors is performed through the used of ASME Code Cases N-629 and N-631and the reference temperature RT_{T0}. For WWER-440 and WWER-1000 equivalent procedures and equations were developed.

- *Check of Master Curve Application to Embrittled RPVs of WWER Type Reactors, M. Brumovsky, paper included in the International Journal of Pressure Vessels and Piping 79 (2002) 715-721*

The paper discusses the use of real fracture toughness data for integrity assessment of WWER type reactor pressure vessels. A comparison is performed with results obtained using design fracture toughness curves based on Charpy impact data. Since the Master Curve approach was developed mainly for PWR type materials, it is recommended in the paper to check its validity also in WWER type materials based on large data base of results. Comparison of transition temperatures T_k and T_0 for different irradiation conditions of base metal shows that there is practically a linear dependence between these two transition temperatures, but transition temperature T_0 is increasing faster than temperature T_k.

- *Measurement of fracture toughness transition behaviour of Cr–Ni–Mo–V pressure vessel steel using pre-cracked Charpy specimens, M. Holzmanna, I. Dlouhy´and M. Brumovsky, Paper included in the International Journal of Pressure Vessels and Piping 76 (1999) 591-598.*

The potential application of small pre-cracked Charpy specimens for the prediction of the fracture toughness of the 1T-thickness specimens and the reference temperature T0 is examined in this paper.

The fracture toughness transition region of small pre-cracked specimens was shifted to lower temperatures as compared with that of 1T SENB specimens. The fracture toughness data of small pre-cracked specimens were size corrected to 1T thickness and used to establish the reference temperature T0 and KJc(mean) fracture toughness curve. The calculated temperature T0 was in consistence with that of the 1T SENB specimen. However, some corrected fracture toughness data lay outside the scatter band of 1T thickness specimens and the shape of the KJc(mean) curve has been quite different from the KJc(med)(1T) curve. Bearing in mind the work of Koppenhoefer and Dodds, and the most recent analysis of Ruggieri, fracture data of small pre-cracked specimens having the validity parameter lower than 50 were first constraint adjusted using the cleavage fracture toughness scaling model of Dodds and co-workers, and only then size corrected. Using this procedure the KJc(mean) curve of such treated data was identical with KJc(med)(1T).

- *Fracture toughness of 15X2MFA steel and its weldments, B.T. Timofeev, G.P. Karzov, A.A. Blumin, V.V. Anikovsky, Paper included in the International Journal of Pressure Vessels and Piping 77 (2000) 41-52*

The paper gives fracture toughness test results of 15X2MFA steel (10 heats), 15X2MFAA steel (six heats) as well as their welds. All these materials are widely used in the manufacture and repair of the WWER-440 reactor pressure vessels, the main elements of which are made from 15X2MFAA (shells of the core zone) and 15X2MFA steel (flange, shell of nozzle zone, bottom). The obtained information is a reliable basis for the construction of reference fracture toughness temperature dependences for these materials. The experimental data of the base metal were obtained for different crack orientations: along and across deformation direction of base metal. Altogether 231 test results were obtained on K1c and KJC criteria. For welds, produced by using various welding methods, about 200 values of fracture toughness were obtained. These tests were performed on small and large CT and SENB specimens, and crack orientation was in conformity with both length and thickness of weld.

- *Determination of crack arrest toughness for Russian light water reactor pressure vessel materials, B.T. Timofeev, G.P. Karzov, A.A. Blumin, V.I. Smirnov, Paper included in the International Journal of Pressure Vessels and Piping 77 (2000) 519-529.*

The experimental results of crack arrest fracture toughness for 15X2MFA and 15X2NMFA steels and their welds, produced by submerged arc welding (SAW) are presented in this paper. The data base containing 176 test results has been obtained for materials used in the manufacture of the WWER-440 and WWER-1000 reactor pressure vessels in the as-produced condition (BOL) and after embrittlement simulated by heat treatment, corresponding to the material state at the end of life (EOL).

- *Fracture Toughness of 10GN2MFA Steel and its Welds, B.T. Timofeev, A.A. Blumin and V.V. Anikovsky, Paper included in the International Journal of Pressure Vessels and Piping 77 (2000) 195-201*

Experimental results on fracture toughness crack initiation for 10GN2MFA steel and its weldments are presented in this paper. The data array includes more than 200 test results. This steel is widely used in nuclear power engineering for pressure vessel, steam generators, collectors and piping of the main cooling system of NPP with WWER-1000 reactors. It has been shown that the fracture toughness depends considerably on the type of base metal (forging, plate, tube) and the steel melting process (OHF, ESM). Welded joints do not have an inferior brittle fracture resistance compared with base metal. However, there are few experimental data for welds and it will be necessary to continue the accumulation of experimental data.

- *The Euratom 5th Framework Programme Project FRAME. Description of the Project and First Results, M. Valo, K. Wallin, E. Lucon, M. Kytka, M. Brumovsky, B. Acosta, L. Debarberis, J. Kohopää, F. Gillemot, M. Horvath, Paper included in the International Nuclear Engineering and Design 235 (2005) 445-455*

The objectives, the experimental research programme and the first results of the FRAME project are described in this paper. Altogether 26 different materials were irradiated in a single irradiation capsule in the HFR in Petten, and the Master Curve based T_0 shifts were measured for each material. In addition four materials were studied in the programme extension. Altogether approximately 700 fracture toughness specimens were tested in the project. The aim of FRAME was to start systematic development of Master Curve based embrittlement monitoring. No bias was noticed between the three main testing project partners in the created database. A first estimate for a model, which describes the dependence of embrittlement on the copper, phosphorus and nickel contents of the material, was given.

- *Master Curve Evaluation of Irradiated Russian WWER Type Reactor Pressure Vessel Steels, H.W. Viehrig, J. Boehmert, J. Dzugan and H. Richter, Paper presented on the 20th International Symposium on Effects of Radiation on Materials, ASTM STP 1405, 2001*

This paper presents results of a joint German/Russian irradiation program performed on the Unit 2 of the Rheinsberg NPP. The experiment comprises Charpy V-notch (CVN), precracked Charpy size (SENB) and compact tension (CT) specimens made of different heats of Russian WWER type reactor pressure vessel (RPV) base and weld metals. References temperatures, T_0, were evaluated according to the Master Curve concept using the multi temperature method. Neutron irradiation induced ductile-to-brittle transition temperature (DBTT) shifts determined on the basis of CVN and SENB tests were compared. On the base of the DBTT the neutron embrittlement sensitivity and the annealing behaviour of tested RPV steels were evaluated. Different heats of the same WWER-RPV steel exhibit different neutron induced embrittlement and annealing behaviour.

8 Microstructure (Mechanisms and micro structural evolution)

Session Summary

In 10 reviewed papers the actual problems and findings in the area “microstructure” were discussed. Irradiation damage (neutron embrittlement) of WWER steels is the most discussed mechanism leading to material degradation during NPP operational time. Different testing methods were used in different laboratories during more than 20 years period. In many cases well balanced combination of western and eastern scientists focused the common effort on topics as:

- Role of selected alloying elements on neutron embrittlement
- Application of rare and very précised testing methods (SANS, APFIM, PAS, TEM, MS, …) for micro structural study of steels
- Comparison of results form HV10 and Doppler broadening S, TEM and PAS, PAS and SANS, etc. and these results to findings from destructive tests

Consolidated Conclusions

It was confirmed that only the proper combination of testing methods can bring benefit in form of new knowledge. Tensile and Charpy impact tests, from which the ductile to brittle transition temperature (DBTT) can be calculated, are for more than 40 years the scientific base for evaluation of material degradation (Debarberis). Comparison of results from destructive and non-destructive testing methods has to be based on deep theoretical knowledge about material microstructure (Boehmert, Miller, Popp). Reviewed papers were focused on the commercially used reactor pressure vessel steel. In conclusion several authors stressed attention on necessity to use computer simulation approach for prediction of neutron embrittlement. Approach towards deep study of binary alloys (or 3-nary) with the aim to reveal the role of most important alloying elements like C, Cu, P, or Ni was mentioned and recommended, too.

The conclusions of the papers were confirmed by the participants.

Open Issues

The most important challenge for the future is ensuring the safe operation behind the projected lifetime. Effective lifetime management based on a clear methodology for transparent and knowledge based approaches will be crucial in the future for western as well as WWER reactors. This challenge is surely worth for a next deep study. Review papers can significantly help the next generation of scientists. A huge amount of experimental results can be used very effectively by verification of different computer models directed to proper simulation of defects creation in RPV steels.

Reviewed papers and summaries

- *Correlation between irradiation-induced changes of microstructural parameters and mechanical properties of RPV steels, Boehmert J., Viehrig H.-W., Ulbricht A., Paper was published in Journal of Nuclear Materials, 334, (2004), p.71-78*

Radiation hardening, displayed by the yield stress increase, and irradiation embrittlement, described by the Charpy transition temperature shift, were experimentally determined for a broad variety of irradiation specimens machined from different reactor pressure vessel base and weld materials and irradiated in several WWER-type reactors. Additionally, the same specimens were investigated by small angle neutron scattering. The analysis of the neutron scattering data suggests the presence of nano-scaled irradiation defects. The volume fraction of these defects depends on the neutron .fluence and the material. Both irradiation hardening and irradiation embrittlement correlate linearly with the square root of the defect volume fraction. However, a generally valid proportionality is only a rough approximation.

In detail, chemical composition and technological pretreatment clearly affect the correlation.

- *Kozloduy NPP Unit # 1 pressure vessel boat sampling, Cvitanovic M., Krpanec G., Paper was presented at International Conference in Prague, 8.-10.12.1997*

The Kozloduy NPP, unit 1, Reactor Pressure Vessel has no reactor vessel material surveillance program. Changes in the material fracture toughness resulting from fast neutron irradiation can not be monitored without removal of material from the vessel itself. On the weld No.4 which is exposed to maximum neutron flux three main and three spare locations for boat sampling are determined.

Preceding to boat sampling from RPV wall, ultrasonic base metal wall thickness was taken, and also weld centerline and the form of weld crown were found by replica method on all sample sites. On the basis mini-Charpy specimens which are to be cut from templates are modeled. Special magnets are placed on the RPV wall and they secure precise positioning on sample locations.

Boat sampling is performed by electrical discharge machining and after this the depth of the divot left on the RPV wall is also measured by replica method.

To minimize stress concentration effects in the divot and to eliminate any surface irregularities resulting from EDM cutting process, the sample areas have to be ground to a radius of 300 mm. The grinding is also implemented to eliminate the recast layer at the surface of the EDM cut. Ultrasonic, liquid penetrant, magnetic particles and visual examinations of the divot sites after grinding were performed.

- *Studies of radiation embrittlement of model alloys by positron annihilation, thermo-electric and magnetic measurements, Debarberis L., Acosta B., Sevini F., Pirfo S., Hyde J.M., Hutchings M.T., Ortner S., Paper was published in NTD&E International 37 (2004) 19-22*

The most important effect of the degradation by radiation in steels and welds is the decrease in their ductility. The main ways to determine the mechanical behaviour of such steels is tensile and Charpy impact tests, from which the ductile to brittle transition temperature and its increase due to neutron irradiation can be calculated. These tests are destructive and regularly applied to assess the integrity of structural materials. The possibility of applying validated non-destructive monitoring techniques would however facilitate the surveillance of such materials.

In this paper, the positron annihilation line-shape analysis performed on a limited number of complex model alloys with different chemical composition is presented. The results are compared with the values previously obtained by thermo-electric and magnetic Barkhausen measurements and with the results of Charpy impact destructive testing. The obtained results prove that positron annihilation is a good indicator of the change in material properties, given that its main parameter increases by effect of radiation as it occurs with the ductile to brittle transition temperature, and the magnetic and thermoelectric properties.

- *Multi-component clustering in WWER-type pressure vessel steels - thermodynamic aspects and impact on SANS, Gokhman A., Boehmert J., Ulbricht A., Paper was published in Journal of Nuclear Materials, 334, (2004), p.195-199*

Atom probe ion microscopic investigations of irradiated WWER 440-type reactor pressure vessel steels suggest the appearance of multi-component clusters. Numerical calculations of the negative minimum of the thermodynamic driving forces for a multi-component system are carried out considering a quasi-quaternary system consisting of Fe, Mn, Si and vacancies. A relative minimum was only found for a composite model of the multi-component clusters composed of a Fe containing core and a vacancies-rich shell. The ratio of their sizes is estimated from the condition that such structures agree with the small angle neutron scattering (SANS) curves measured.

- *Comparison of microstructural features of radiation embrittlement of WWER-440 and WWER-1000 reactor pressure vessel steels, Kuleshova E.A., Gurovich B.A., Shtrombakh Ya.I., Erak D.Yu., Lavrenchuk O.V., Paper was published in Journal of Nuclear Materials, 300 (2002), p.127-140*

Comparative microstructural studies of both surveillance specimens and reactor pressure vessel (RPV) materials of WWER-440 and WWER-1000 light water reactor systems have been carried out, following irradiation to different fast neutron .uences and of the heat treatment for extended periods at the operating temperatures. It is shown that there are several microstructural features in the radiation embrittlement of WWER-1000 steels compared to WWER-440 RPV steels that can cause changes in the contributions of di.erent radiation embrittlement mechanisms for WWER-1000 steel.

- *Microstructurer alterations in the base materials, heat affected zone and weld metal of 440-WWER-reactor pressure vessel caused by high fluence irradiation during long term operation; material:15 Ch2MFA = 0.15 C-2.5 Cr-0.7 Mo-0.3V, Maussner G., Scharf L., Langer R., Gurovich B., Paper was published in Nuclear Engineering and Design, 193, (1999), p.359-376*

Within the scope of the EC research project Tacis ’91 (‘RPV-Embrittlement’), trepans were taken from the highly irradiated circumferential RPV-weld of the Novovoronesh power plant unit-2 of the type WWER-440:230. The cumulated fast fluence level in this position reaches up to 6.5_1019:cm2 (*E*\0.5 MeV). In a joint research work, the mechanical properties, the chemical composition, and the microstructure of the base material, the heat affected zone (HAZ), and the weld metal have been investigated in order to study the influence of irradiation, and of post irradiation heat treatment (475°C, 560°C) on the properties. The examination of the microstructure performed by analytical transmission electron microscopy (200 kV) shows the existence of dislocation loops (‘black dots’), irradiation induced precipitates, and segregation of copper in the carbides. These changes in microstructure, which are due to service affection (neutron irradiation, temperature) have occurred more pronounced in the weld metal and the Heat Affected Zone than in the base material.

- *Atom probe tomography of 15Kh2MFA Cr-Mo-V steel surveillance specimens, Miller M., Russell K.F., Kocik J., Keilova E., Paper was published in Micron 32, (2001), p.749-755*

An atom probe study has been performed on 15Kh2MFA base and 10KhMFT weld metal surveillance specimens from a WWER-440/213C reactor to investigate the mechanisms that produce embrittlement in low copper materials during service. The composition of the base metal was Fe-0.06 at.% Cu, 3.1% Cr, 0.34% V, 0.46% Mn, 0.35% Mo, 0.07% Ni, 0.34% Si, 0.74% C, 0.025% P, and 0.028% S. The base material was characterized after thermal aging for 10 years at 295^{o}C and after neutron irradiation at 270^{o}C for 10 years to a fluence of 1.0 £ 1025 m22 (E . 0.5 MeV). The ductile-to-brittle transition temperatures (DBTT) of the base metal were 249, 270 and 14^{o}C, for the unirradiated, thermally aged and neutron irradiated conditions, respectively. The composition of the weld metal was Fe-0.05 at.% Cu, 1.46% Cr, 0.22% V, 1.11% Mn, 0.29% Mo, 1.17% Si, 0.17% C, 0.02% P, and 0.029% S. The weld material was characterized after tempering for 18h at 690^{o}C plus a simulated stress relief treatment of 43.5 h at 680^{o}C, after thermal aging for 5 years at 295^{o}C, and after neutron irradiation at 275^{o}C for 5 years to a fluence of 5.2 x 1024 m2 (E . 0.5 MeV). The DBTTs were 7, 11 and 123 ^{o}C, respectively, for these three conditions. A high number density of ultrafine manganese- and silicon-enriched regions was observed in both neutron-irradiated materials. Phosphorus segregation was observed at the VC-matrix interface and at grain boundaries.

- *Evaluation of thermal annealing behavior of neutron-irradiated reactor pressure vessel steels using nondestructive test methods, Popp K., Brauer G., W.-D.Leonhardt, H.-W. Viehrig, Paper was published in “Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels. An international review (Third Volume) ASTM SPT 1011, L.E Steele, Ed. American Society for Testing and Materials, Philadelphia, 1989, 188-205*

Usually the assessment of the irradiation sensitivity and annealing behaviour of reactor pressure vessel (RPV) steel is performed by means of destructive test methods, mainly impact and tension tests. In the paper a new kind of search for an efficient temperature-time regime for post-irradiation thermal heat treatment is presented using nondestructive test methods like positron annihilation (Doppler broadening parameter S) and hardness (Vickers hardness HV10).

Samples of Cr-Mo-V RPV steels (Soviet type 15Kh2MFA) were irradiated to different fluence levels of fast neutrons at temperatures T<156°C in a test reactor (base metal) and T=265°C in a pressurized water reactor (base as well as weld metal). From isochronal and isothermal annealing curves of HV 10 and S, favorable temperature-time regimes for each type of irradiated material were estimated. The data obtained from tension and impact tests indicate that sufficiently large recoveries took place by application of these regimes.

The new approach presented is especially useful in such cases where only the smallest amounts of irradiated materials are available - a case often met for RPV surveillance specimens.

- *Investigation of radiation damage in WWER-440 reactor vessel steels by SANS, Saroun J., Kocik J., Strunz P., Published in Physica B350, (2004), e755-e757*

The effect of neutron irradiation on the microstructure of WWER-440 type reactor pressure vessel steels was studied by small-angle neutron scattering (SANS). The SANS spectra were fitted by a model of non-ferromagnetic particles in saturated ferromagnetic matrix, which permitted us to evaluate apparent volume fractions (weighted by scattering contrast), size distributions and mean ratio of magnetic to total scattering cross-sections for small (Ro3 nm) irradiation-induced precipitates. Close correlation between the apparent volume fraction of the damaged domains and ductility transition temperature was observed. Differences in the volume fractions obtained from magnetic and nuclear scattering indicated that chemical composition varied with increasing .fluence and could be explained by increasing concentrations of solute atoms in the damaged domains. This coarsening effect is also manifested on the size distributions by slight growth of mean domain radius.

- *Defects investigation in neutron irradiated reactor steels by positron annihilation, Slugen V., Published at International conference on Structural Mechanics in Reactor Technology (SMiRT17), Prague, Czech Republic, August 17-22, 2003*

Positron annihilation spectroscopy (PAS) based on positron lifetime measurements using the Pulsed Low Energy Positron System (PLEPS) were applied for the investigation of defects of irradiated and thermally treated reactor pressure vessel (RPV) steels. PLEPS results showed that the changes in microstructure of the RPV-steel properties caused by neutron irradiation and post-irradiation heat treatment can be well detected. From the lifetime measurements in the near-surface region (20-550 nm) the defect density in Russian types of RPV-steels was calculated using the diffusion trapping model. The post-irradiation heat treatment studies and the correlation with results from hardness measurements performed on non-irradiated specimens are also presented.

9 Testing (Property-Property Correlation)

<u>Session Summary</u>

17 papers were reviewed under this heading, mainly regarding testing techniques. In six papers the application of miniature specimens was discussed (Valo, van Walle, Wallin, Korolev). Four papers discuss non-destructive evaluation methods (Acosta, Bakirov). Four papers deal with testing and microstructure (Kocik, Kuleshova, Rétfalvi, Slugen). Three papers were of more general character (Laureova, Margolin, Debarberis).

<u>Consolidated Conclusions</u>

In general, the conclusions in the papers were confirmed by the participants.

- There is a relatively good correlation between standard and miniature specimen impact testing, however the average σ= 21°C scatter of the miniature KLST specimens is to large (generally the allowable irradiation transition shift for RPV welds is in the range of 80-120°C. Shift measurement requires at least 2 test sets consequently the testing uncertainty may reaches the 30% of the allowable shift, and reduces the operational lifetime by 50% or more.
- Master Curve: The optimum is to use 10x10 or 5x5 specimens. Kim Wallin calculated, that in case of testing of smaller size specimens the scatter increasing, consequently to reach the same reliability level the required number of specimens is rapidly increases too.
- The non-destructive STEAM technique has been successfully used on aged specimens
- The non-destructive Barkhausen noise technique has some potential to be used
- The Ball indentation technique is a candidate for semi non-destructive testing

- In WWER-440 steels the radiation defects are dislocation loops. The density of rounded copper enriched precipitates and the disk shaped vanadium carbides increases the effect of irradiation. In the WWER-1000 steel the density of rounded copper enriched precipitates increases, while the disk shaped chromium_carbides precipitation density remains stable. At similar fast neutron fluence the density of radiation induced precipitates in WWER-1000 steel is 100-1000 times lower than their density in WWER-440 RPV steels. To evaluate the embrittlement mechanism of the WWER-1000 steel long heat exposition (30.000-60.000 hours at 270-290°C) was applied without irradiation. High nickel content caused the formation of phosphorus segregation and embrittlement. On the other hand, nickel affects the size (decreasing) and the distribution of the Cu rich precipitates. The increasing number of clusters is acting as obstacle to the dislocation motion, resulting hardening and embrittlement.
- There is no need for distinguishing at Master Curve testing of WWER and PWR materials. The standard testing method is equally valid for all low alloyed RPV materials.

Open Issues

- Is the correlation among Charpy and Master Curve testing results the same for “low” and “high” embrittled materials?

Reviewed papers and summaries

- *A preliminary evaluation of irradiation damage in model alloys by electric properties based techniques, B. Acosta, F. Sevini, L. Debarberis, International Journal of Pressure Vessels and Piping 82 (2005) 69-75*

The nondestructive evaluation (NDE) of radiation embrittlement is a developing field of the material testing. Several RPV-s haven’t surveillance program, or the surveillance specimens don’t provide enough information about the material degradation during the lifetime. This is frequently occurs at lifetime extension. Mechanical testing is expensive even if the irradiated archive material is available. The authors applied two electric methods to determine the material degradation: the thermoelectric power measurement (STEAM) and the electric resistivity (REAM) testing. To determine the effects of the chemical alloying 20 different model alloys have been tested and irradiated covering a wide spectrum of Ni, Cu, and P content. Charpy transition temperature shifts have been correlated with the measured electric properties. The STEAM method gave linear correlation with the embrittlement, while the REAM vs. transition shift relation is exponential. The authors combined the two measurements and used the ratio of the two test results. This new parameter gives better description of the material degradation than the single test results.

- *Cu effects on radiation embrittlement of Ni-containing complex model alloys and the related potentials of the thermoelectric method, B. Acosta, L. Debarberis, F. Sevini, A. Kryukov, NDT&E International 37 (2004) 321-324*

The nondestructive method to obtain mechanical properties of the materials (STEAM) is based on the measurement of the thermoelectric voltage generated by the Seebeck and Thomson effect. This technique is used to evaluate the Cu effect on radiation embrittlement. Radiation embrittlement is the sum of three embrittlement mechanism: occur of segregations and precipitations, and matrix damage. The irradiation of the Cu and Ni leads to occur nano-precipitates blocking the movement of dislocations consequently it increases the hardness and reduces the toughness of the material. The reduction of Cu content by precipitation increases the Relative Seebeck Coefficient (RSC). The phosphorus segregates on the grain boundaries doesn’t affect the RSC values even its role in the irradiation embrittlement is significant. The STEAM results helps the understanding the Cu precipitation during irradiation and describe the embrittlement mechanism for model development.

- *A phenomenological method of mechanical properties definition of reactor pressure vessels (RPV) steels WWER according to the ball indentation diagram, M.B. Bakirov, V.V. Potapov, J.-P. Massoud*

Newly developing semi nondestructive method based on intending an instrumented hard ball into the surface of the tested material. Hardness tests are used from long time to evaluate the strength of the metals. Stepwise indentation with instrumented ball indenter allows measuring the load and deformation together. An analytical solution of conversion of the data into uni-axial stress-strain curve has been elaborated and introduced. Based on this solution the flow-curve testing of an operating structure can be performed without cutting specimens and destroy it. From the flow-curve even fracture toughness can be assessed. The presently elaborated methodology and testing device calibrated and validated on a wide range of as received and irradiated RPV steels.

- *Examination of metal condition of the reactor pressure vessel and the core barrel at Rostov NPP Unit 1 by non-destructive magnetic-hardness testing technology, M.B. Bakirov, V.V. Potapov, N.Y. Zabruskov*

Kinetic hardness method consists in recording of the complete diagram of penetration of a ball indentor into the investigated material. The consequent mathematical processing allows from the diagram of penetration to get the standard diagram of monoaxial tension. To further develop the kinetic hardness testing it has been combined with acoustic emission measurement. This method is successfully applied on steel surfaces. The second generation of RPV-s is cladded with a 6-10 mm thick austenitic layer of which prevent to apply ball indentation. MAGNETEST – a new device measuring of magnetic coercive force and Reley coefficient has been developed. The intensive magnetic field penetrates the paramagnetic austenitic cladding and provides information on the mechanical properties of the ferritic base and weld materials. Several tests have been performed to validate the results obtained by magnetic measurement. The magnetic tests results may correlate with the hardness or ball indentation results. Phosphorus segregation at the grain boundaries causes nonhardening embrittlement and can't be evaluated by magnetic testing. The new magnetic testing is a candidate method to be applied at site testing too.

- *WWER-1000 base metal reference steel and its characterisation, L. Debarberis, B. Acosta, S. Pirfo, F. Sevini, A. Kryukov, A. Chernobaeva, F. Gillemot, M. Brumovsky, Transactions of the 17th International conference on Structural Mechanics in Reactor Technology (SMIRT-17) Prague, 17-22 Aug, 2003*

To study new testing methods and to compare irradiations in different type research and power reactors well characterized reference steels are required. Homogeneity and a certain level of radiation sensitivity are also requirements. The IAEA JRQ and JFL the US HSST-01, 02, 03 plates and the ASTM A302B standard reference material are not representative of the materials of the WWER reactors. JRC-IE and Kurchatov Institute obtained a trepan cut from an industrially manufactured nozzle ring of a WWER-1000 reactor. This material contents medium amount of Ni and Cu, the sensitivity is ideal to use as reference material for irradiation monitoring.

A test programme has been elaborated to study the properties in axial and in radial directions, to evaluate the homogeneity. Charpy impact test performed at Kurchatov Institute and at JRC-IE, tensile testing performed at the Ukrainian Nuclear Research Institute. A non-destructive evaluation of the mechanical properties was also performed by using the STEAM method (thermoelectric voltage) and the Barkhausen noise. The results show that the trepan satisfies the requirements of a reference steel.

- *TEM and PAS study of neutron irradiated WWER-type RPV steels, J. Kocik, E. Keilova, J. Cízek, I. Prohászka, Journal of Nuclear Materials 303 (2002) 52-64*

The knowledge of the microstructure of the reactor materials in as received and irradiation aged state is essential for successful modelling and development of new irradiation resistance steels. Surveillance specimens from WWER-440 and WWER-1000 RPV materials and irradiated A533B steel have been studied. Two special heats of the WWER-1000 reactor base material also produced with increased Cu and P pollution. The materials were irradiated in surveillance channels of WWER-440 reactor and in the LVR-15 research reactor. Two sensitive methods used for study of the ageing caused changes in the microstructure:

- TEM (Transition Electron Microscopy) used to study the dislocation structure
- PAS (Positron Annihilation Spectroscopy) positron lifetime (PL) and Doppler broadening (DB) used to study the vacancy clusters.

The results shown three kinds of neutron radiation induced matrix defects in power reactor surveillance specimens. The fine precipitates concentrated near the dislocation in the materials irradiated in research reactor. No voids were found. Only point defects discovered in the research reactor irradiated samples showing that the irradiation time were short for occurring of dislocation loops.

- *Problems of underclad type defects in reactor pressure vessel integrity evaluation, D. Lauerova, M. Brumovsky, P. Sunpanen, J. Kohopää, Transactions of SMIRT 17, Prague, 17-22 August 2003 paper G02-2*

Underclad defects become important at calculation of RPV lifetime of the WWER Reactors. Large size TPB (three point bending) specimens (70*100*670) were used to study the behaviour of the underclad defects. Seven tests performed on specimens which were cladded after cracking (subclad crack production). Semi elliptical and through the specimen widht craks applied. To study the effect of the base material properties on the crack propagation 3 specimens are aged thermally. The behaviour of the specimens is modelled by 2 and 3 D finite element models. During the tests some pop-in occured. Modelling and fractography testing shown, that the cracks mostly propagated into the base material, and arrested in the clad, even if the deformation energy (J) were larger at the crack front at cladding than in the base material side.

- *The ductile crack growth effect on the temperature dependence of cleavage fracture toughness, B. Z. Margolin, V.I. Kostylev, and A. I. Munkin, Transactions of SMIRT 17, Prague, 17-22 August 2003 paper G03-3*

The cleavage fracture toughness at low temperature can be characterised by an exponential curve like the Master Curve. At increased temperature ductile deformation at the crack tip occurs, and after certain deformation cleavage fracture occurs. The Master Curve is valid until the temperature range of the large scale deformation, and became very conservative when the cleavage fracture occurs after plastic deformation. A new model has been elaborated for modelling cleavage fracture after ductile crtack growth. It uses probabilistic model for cleavage fracture and deterministic model for ductile fracture. The results show that in case of highly embrittled materials in the upper temperature range the Master Curve overestimates the real fracture toughness, while in the case of initial material conditions it may be too conservative. The new suggested curve provides solution for these difficulties. Since the deviation of the two curves is only in the upper end (where the Master Curve is not valid) the problem may occur only at very highly embrittled steels. In this range the new curve may gives better results. The theory has been verified on irradiated WWER base and weld materials.

- *Applicability of miniature size bend specimens to determine the master curve reference temperature T0, K. Wallin, T. Planman, M. Valo, R. Rintamaa, Engineering Fracture Mechanics 68 (2001) 1265-1296*

Master Curve testing can be performed on small size specimens, but the use of different size and type specimens may cause increased scatter or bias of the results. It has been shown that the T stress in the CT and TPB specimens are different, consequently the CT specimens provide 8-10°C more conservative T_o values than the TBP specimens. Use of smaller specimens increases the scatter of the results. Very small specimens can't be considered homogeneous ones, since even microscopic defects are relative large compared with the specimen size. Several experimental tests have been performed on WWER base and weld materials as well as on JRQ and on American A533B and A508 steels in as received and irradiated conditions. The following conclusions determined:

- The small size specimens can be used for Master Curve testing in the temperature range -50<T-To <-20°C.
- The optimum material volume is used for 5*5 mm specimens
- There is about 8°C bias between the CT and TPB specimens due to the different T stresses, but it isn't large enough to use corrections.

- *Characterisation of the mechanical properties of the WWER 440 weld No.502 with small specimens, M. Valo, R. Ahlstrand, Y. Kohopää, K. Wallin, IAEA Specialists Meeting on Irradiation Embrittlement and Mitigation, 26-29 April 1999 Madrid*

Small size specimen use is required when the available amount of archive material is limited or when trepans cut from operating structures (e.g. RPV) to evaluate the ageing rate. In the frame of the IAEA Co-ordinated Research Programme „Round-Robin exercise on WWER-440 RPV weld metal irradiation embrittlement, annealing and re-embrittlement" VTT used different small size specimens and compared the results. Charpy V notched impact and TPB (three point bend) static fracture toughness specimens were used with cross sectional dimensions 10*10 mm, 5*5mm and 3*4 mm. The specimens irradiated in the Loviisa reactor for 3 years. The results have shown that the Master Curve To values measured on small size specimens after size adjustment correlate well with the Charpy size TPB specimen results. The as-received material Charpy transition temperatures are within an 30°C range. Irradiation embrittlement measured with small specimens has a tendency to underestimate the effect of irradiation i.e. small specimens give non-conservative estimates.

- *Reconstitution techniques qualification and evaluation to study ageing phenomena of nuclear pressure vessel materials (RESQUE), E. Van Walle, M. Scibetta, M.J. Valo, H-W. Viehrig, H. Richter, T. Atkins, M.R. Wootton, E. Keim, L. Debarberis, M. Horsten, IAEA Specialists Meeting on Irradiation Embrittlement and Mitigation, 26-29 April 1999 Madrid*

The EU 5th Framework project RESQUE elaborated a technology for reconstituting Charpy size specimens by stud welding. Eight different European institutes co-operated in the frame of the research. Archive material not frequently available, especially in the case of irradiated RPV surveillance specimens. The remnants of the archive specimens can be used to manufacture 2-4 times more reconstituted specimens as the original programme had. A careful technology has been elaborated for the reconstitution. The effect of the heat input studied and optimum parameters are selected. The reconstituted specimens tested by three point bending and X-ray tests. Metallography and hardness tests used to determine the heat affected zone. A technology for machining the welded specimens was also developed, providing minimum deviation from the standardised sizes. Test series have been performed on reconstituted and original as received and irradiated Charpy V impact and three-point bend static fracture specimens to validate the use of the technology.

- *Positron annihilation and Mossbauer spectroscopy applied to WWER-1000 RPV steels in the frame of IAEA High Ni Co-ordinated Research Programme, V. Slugen, A. Zeman, J. Lipka, L. Debarberis, NDT&E International 37 (2004) 651-661*

During irradiation defects are formed from vacancies and interstitials created in collision cascaded processes. Those point defects surviving the cascades migrate freely through crystal lattice, interacting with each other and with solute atoms in the matrix and also with dislocation substructure and precipitates. These irradiation induced diffusion processes result in the formation of new defect clusters, dislocation loops, and precipitates. Positron annihilation spectroscopy (PAS) and Mösbauer spectroscopy (MS) were applied the microstructure and degradation processes of elevated Ni content WWER-1000 steels. Special interest was given to the synergetic mechanism of Ni and Cu. The high sensitivity of the PAS and MS parameters allowed to study this interaction. It confirmed the hypothesis that nickel affects the size (decreasing) and the distribution of the Cu rich precipitates. The increasing number of clusters acting as obstacles to the dislocation motion, and it results hardening and embrittlement. Ni contributing this process, and even some clusters enriched in P.

- *Radiation damage study using small-angle neutron scattering, E. Rétfalvi, Gy. Török, L. Rosta, Physica B 276-278 (2000) 843-844*

Small angle neutron scattering is a useful tool to study the change of nanostructures. Knowledge on nanostructure changes during irradiation is the base of modelling. Irradiated and non-irradiated WWER 440 base material and weldment was studied. Tests were performed at the Budapest Neutron Center. 1.2 T external magnetic field perpendicular to the neutron beam was applied. Fourier transformation performed on measured spectra. The study concluded, that two mean radii defects are occurred during irradiation. The larger ones are phase transformations or melt objects, the smaller ones are precipitates.

- *The actual properties of WWER-440 reactor pressure vessel materials obtained by impact tests of subsize specimens fabricated our of samples taken from the RPV, Yu. Korolev, A. Kryukov, R. Langer, C. Leitz, V. Nikolaev, Yu. Nikolaev, P. Platonov, C.Y. Rieg and Ya. Shtrombakh, Small Test Specimen Techniques ASTM 1329 1997*
- *Assessment of irradiation response of WWER-440 welds using samples taken from Novoronezh unit 3 and 4 reactor pressure vessels, Yu. Korolev, A. Kryukov, Yu. Nikolaev, P. Platonov, Ya. Shtrombakh, R. Langer, C. Leitz, C.Y. Rieg, Nuclear Engineering and Design 185 (1998) 309-317*
- *The properties of WWER-440 type reactor pressure vessel steels cut out from operated units, Yu. Korolev, A. Kryukov, Yu. Nikolaev, P. Platonov, Ya. Shtrombakh, R. Langer, C. Leitz, C.Y. Rieg, V. Nikolaev, Nuclear Engineering and Design 195 (2000) 137-142*

The use of Subsize specimens is the solution of testing trepans cut from operating RPV-s. Novovoronyez unit 2, 3 and Kozloduj unit 2 have been annealed in the 90's due to the high irradiation embrittlement of the welds.

Previously the Subsize Charpy specimens were not standardized. Several formulae were used to convert the subsize Charpy results into standard DBTT (ductile brittle transition temperature obtained on standard 10*10 cross section Charpy V impact specimens).

Based on experimental work the following formula elaborated for the conversion:

$$T_k^{10*10} = 50 + T_k^{5*5}$$

The scatter of the conversion is ±42°C

Based on the results of the trepans annealing effects were evaluated. Before annealing the DBTT temperature reached nearly 200°C at unit 3 and 4 of Novovoronyez. The annealing at 475°C for 100 hours resulted in a DBTT with only about 20°C residual shift. Previously 430°C at 100 hours annealing was applied on unit 3 Novovoronyez, where the residual DBTT shift was about 50°C.

After irradiation, and annealing the re-irradiation shift was lower than expected with the previously elaborated conservative or lateral model.

- *Comparison of microstructural features of radiation embrittlement of WWER-440 and WWER-1000 reactor pressure vessel steels, E.A Kuleshova, B.A. Gurovich, Ya. I. Shtrombakh, D. Yu. Erak, O.V. Lavrenchuk, Journal of Nuclear Materials 300 (2002) 127-140*

Understanding of the radiation emkbrittlement mechanisms enhance the future RPV steels. WWER-440 and WWER-1000 steels mainly differ only in the content of Ni, but the irradiation embrittlement rate is quite different. Microstrutural studies have been performed to evaluate these differences.

Different microstructural tests were performed on the selected steels in received and in irradiated state: fractography, X-ray microanalyser, and transmission electron microscopy. The results have been compared with the results of mechanical testing. Samples also thermally aged for 30000-60000 hours to separate the thermal ageing effect from the irradiation embrittlement.

The results shown that in WWER-440 steels the radiation defects are dislocation loops. The density of rounded copper enriched precipitates and the disk shaped vanadium carbides increased on the effect of irradiation.

In the WWER-1000 steel the density of rounded copper enriched precipitates increased, while the disk shaped chromium carbides precipitation density remained stable. At similar value of fast neutron fluence the density of radiation induced precipitates in WWER-1000 steel is 100-1000 times lower than their density in WWER-440 RPV steels. To evaluate the embrittlement mechanism of the WWER-1000 steel long heat exposition (30000-60000 hours at 270-290°C) applied without irradiation. High nickel content caused the formation of phosphorus segregation and embrittlement.

10 PLEX (PLEX Issues)

<u>Session Summary</u>

In this study totally 10 papers have been reviewed. The selected papers cover mainly PLIM (Plant Life Management) and PLEX (Plant Life Extension) activities of WWER NPPs in Russia and other WWER owner countries (Czech Republic, Hungary, Bulgaria, Slovakia and Finland). Some of the reviewed papers are rather comprehensive and give detailed technical information regarding PLIM and PLEX. On the other hand, some papers are describing the issue on a rather general level and no technical details are given. Most of the reviewed papers are written by technical research institutes, some papers by Safety authorities and utilities.

In Russia currently 4 WWER 440 NPPs of 1st generation V-230 type have reached the end of their design life (30 years). The life time of these units have been extended; first by 5 years and later even more. In Finland the life time of Loviisa 1, which is a 2nd generation WWER 440 unit of type V-213, has been extended by 15 years. However, the life time of the RPV of Loviisa 1 is under special surveillance and need re-licensing more frequently during the PLEX time due to neutron irradiation embrittlement of the core weld.

In the Russian units where PLEX has been implemented, huge and expensive modernization programs have been realized before the end of design life of each unit. Neutron embrittlement and consequently the integrity of the RPV of the WWER 440 NPPs is seen as the most important and critical issue. The RPV is considered a non-replicable component. A lot of work and research has been carried out in order to ensure integrity of the RPV core weld both before and during the PLEX period. Since the 1st generation units did not have any surveillance program for the RPVs, samples have been cut out of the core weld and the material condition has been determined by using sub size test specimens. Other components can be replaced more easily and economically and the safe lifetime can be managed according to normal PLIM and PLEX procedures.

Consolidated Conclusions

From the review it is obvious, that the neutron embrittlement of the core weld of the 1st generation WWER 440 reactors is the critical issue when considering safe operation and PLEX. A lot of measures have been taken in order to ensure the integrity of the RPV of these NPPs (Brumovsky). The WWER 440 is quite an adaptive NPP providing possibilities for implementation of mentioned measures and consequently prolonged life time for ageing units. According to the Russian Regulatory Authorities (GAN) special attention must still be paid to solve the problem of re-embrittlement of the RPVs after annealing. It has been recommended to carry out extended research on trepans cut out from the decommissioned reactors in Greifswald (Borodkin) in order to have real test results and reduce uncertainties of modelling and correlation on which present RPV licensees for 1st generation WWERs are based on.

The primary piping of these NPPs is made of austenitic stainless steel as mentioned earlier. According to reviewed papers there is practically no effect of thermal ageing, corrosion and fatigue on the integrity of the piping during even a period of 45 years of operation. The probability of a primary pipe break of 250 – 500 mm diameter pipe is very low. According to Russian scientists the DBA could very well be restricted to a break of a 100 mm diameter piping as in the design bases for the 1st generation WWER 440 units. The steam generators of the WWER 440 NPPs have been performing pretty well and no replacements have been necessary so far. In the WWER 1000 NPPs a lot of steam generators have been replaced due to cracking primary collectors. Erosion corrosion or flaw assisted corrosion of the feed water piping in the secondary piping of the WWER 440 NPPs has been a serious ageing problem. This problem can be managed by proper NDT and pipe replacement. Special computer programs are nowadays available for managing this difficult problem properly.

Open Issues

- Correlation of real and sub sized testing
- Re-embrittlement

Reviewed papers and summaries

- *Results on research templates from Kozloduy-1 reactor pressure vessel, P. Platonov, Ja. Strombach, A. Kryukov, B. Gurovich, Ju. Korolev, J. Shmidt, Paper form Nuclear Engineering and design, 191, 1999*

The paper describes activities related to the sampling and testing of templates cut out from the core weld of RPV of Kozloduy-1. Chemical analyses, hardness measurements, microstructure investigation, fluence determination and testing of mini Charpy specimens (mCV) were carried out. The test results show that the re-embrittlement of the core weld after annealing is much smaller than would be expected by applying evaluation methods accepted as the main Russian Standard approach. The aim of the study was to determine the condition of the weld metal in the present condition IAR (irradiation anneal re-irradiation) and to evaluate the integrity of the RPV.

The main and most important results were received in testing the mCV specimens. The basic "as received" condition of the samples were IAR (irradiation anneal re-irradiation). One part of the specimens was heat treated in 475 ^{o}C and another part in 560 ^{o}C in order to simulate the condition of the material after annealing and in the initial state. The results show, that the present toughness properties of the core weld would be considerably lower (86 ^{o}C) than when evaluated theoretically according to the Russian Standard proposal. It also shows that the re-irradiation would be much lower than when evaluated according to the conservative and lateral approach.

However, there is a lot of speculation in above findings. Actually only one point (IAR) corresponds to a real material condition (as sampled condition). The aim of mentioned heat treatments is to simulate the initial as well as the annealed conditions of the weld. There is not enough evidence that the simulation is fully reliable. Furthermore the correlation between the mCV and standard CV testing is not very good. The standard deviation is still very large so it could be dangerous to make too serious conclusions regarding PLEX based on the result obtained in this paper.

- *Examination of WWER-440 RPV steel re-irradiation behaviour using materials from operating units", Ya. Strombach, RRCKI, Journal of Pressure Vessels and Piping 77 (2000)*

The paper is dealing with re-embrittlement kinetics of WWER 440 type 230 RPVs after thermal annealing. Tests have been carried out on samples which were cut out of the inner wall of the core region of some annealed WWER-440 RPVs (Kozloduy 1 and 2, Novovoronesh 2, 3 and 4). From cut out samples mini CV (mCV) specimens were made. Furthermore full size and mCV Charpy-V specimens were produced from trepans of Novovoronesh 2 RPV. The aim was at confirming the correlation between the shifts in DBTT received from mCV and full size CV testing. Re-irradiation kinetics of the core weld material was evaluated based on mCV test results and above correlation. The elaborated re-irradiation properties were compared to procedures given in Russian Guide.

The results of the studies confirmed the old correlation between mCV and standard CV DBTT shifts as adequate. The standard deviation is, however, still rather large ($2\sigma = 50\ ^oC$). The results also showed that the trend curve for primary shift in DBTT is non-conservative and a corrected trend curve was proposed. Furthermore the test results indicated that the conservative- and lateral shifts in re-embrittlement would be conservative based on mCV tests and the above correlation with full size specimens, but the vertical shift is not usable. The lateral shift was proposed as the best one for evaluation of re-embrittlement after annealing.

Some results on flux effect on irradiation embrittlement kinetics were also presented. Test specimens were irradiated in the surveillance positions of WWER 440 reactors with full and reduced fuel loading. It was shown that DBTT shift would be much bigger in low flux environment (lead factor about 6). This finding would mean that surveillance results are non-conservative compared to RPV wall position of the core weld due to a big lead factor (>16).

- *Analysis of justification for extension of operation of Russian NPP with WWER in aspect of pressure vessel radiation Embrittlement, G. Borodkin, I.v. Kaliberda, N.N. Khrennikov, SEC NRS of Gosatomnadzor, ICONE 12, 2004*

In the paper the view of the Russian Regulatory Authority (RA) concerning PLEX of first generation WWER 440 units is described. It is underlined that irradiation embrittlement of the RPV is the key issue when considering PLIM and PLEX of these NPPs. PLEX licence has already been issued for 3 units Russia (Novovoronesh 3 and 4 and Kola 1). In all cases the Russian RA has approved a much shorter PLEX time than applied for by the utility Rosenergoatom.

The basic criteria for RPV integrity assessment and life time evaluation are the Russian Standard PNAE G-7-002086 and the Regulatory Guide RB-007-99. The important issue is to calculate the key property parameter T_k^a, which is the highest acceptable DBTT T_k for the object under consideration. The following measures are recommended in order to ensure or extend service life for the RPV:

- Decrease the fluence rate on the RPV wall (dummy elements in core periphery)
- Softening of PTS by managing processes related to ECCS (increase of ECCS water temperature etc.)
- Thermal annealing of the core region weld

Regarding PLEX for the 1st generation WWER-440 reactors the Russian RA have given special attention to the following issues:

- reliability of recovery in annealing (Tko versus Tka)
- reliability of sub-size specimens results
- reliability of correlation between sub size and standard size Charpy specimen
- re-irradiation kinetics after annealing

The Russian Authorities GAN strongly recommends carrying out extended research on trepans which should be cut out of the decommissioned WWER 440 units in Greifswald. This would give possibility of having real test results and reduce uncertainties of modelling and correlation on which present RPV licensees for 1st generation WWERs are based on. This issue could be of common interest for all WWER-440 owners as well as the EC. Investigation of trepans from Greifswald was planned in the PDS for the present Tareg embrittlement projects (budget year 2000 and 2003), but were excluded from TOR on demand from the Russian side. In those days the trepanning was a controversial and considered too expensive (licensing of cutting procedures etc.). The situation has changed: Trepans have been cut out of the RPVs at Units 1 and 2 and will be easily available for research. This would be the proper time for the main designer Gidropress, the Russian utility REA, the European Commission and others to re-initiate research activities on the WWER 440 RPV embrittlement topic in order to improve the knowledge regarding RPV integrity and lifetime.

- *Material property degradation assessment of the first generation WWER 440 RPV after prolonged operatio", I. Gorinin, B. Timofejev, T. Chernaenko, ZNIIKM Prometey, ST. Petersburg, SMIRT 17, 2003*

In this paper ZNIIKM Prometey's view on PLIM and PLEX of the 1st generation WWER 440 NPPs is described. ZNIIKM was responsible for the material selection and especially welding technology for these type of plants. The paper describes the influence of thermal ageing, cyclic loading and corrosion on the material properties during long term operation. The influence of neutron fluence is not a part of this paper, even though the problem is mentioned in the paper.

The influence of thermal ageing on the RPV material has been studied by accelerated methods by keeping samples in a higher temperature than the operation temperature of the reactor. The ageing temperature was 350 ^{o}C, while the normal operation temperature does not exceed 295 ^{o}C. Furthermore samples were cut out from the primary circuit piping of operating plants after 100 000 hours of operation. The primary piping is made of austenitic stainless steel of type A 321 (08X18H10T). The results showed that the influence of thermal ageing on the toughness properties is negligible and need not to be taken into account when considering PLIM and PLEX of the 1st generation WWER 440 units.

Tests were also carried out in order to investigate the influence of cyclic loading on the toughness properties of the RPV and primary piping materials. According to the test results the cyclic loading did not cause any changes in the DBTT of the materials. In a similar way it was demonstrated that there is no risk of significant general or local corrosion in the RPV or piping during service or extended service life of the RPV or primary piping.

Accordingly thermal ageing, cyclic loading or corrosion of the primary components and piping do not restrict PLIM or PLEX of the 1st generation WWER 440 units. The plant life can well be extended up to 45 years operation when considering the influence of above ageing phenomena according to the paper.

- *Irradiation embrittlement and mitigation, R. Rantala, M. Ojanen, STUK, IWG-LMNPP, Vol 2, Vienna 1995*

The paper describes the situation in Loviisa 1 NPP before the annealing of the RPV core weld in mid 1990 including qualification procedures requested for annealing by STUK. STUK has traditionally issued operating licences for the RPVs in Loviisa NPP for shorter periods since the faster than expected embrittlement of the core weld was discovered in the year 1980 in Finland. The integrity assurance of the RPV in Loviisa NPP was based on deterministic analyses, supported by probabilistic evaluations. The acceptance criteria for deterministic analyses was no crack initiation except for low pressure transients, were benefit from crack arrest was accepted. In the Finnish PTS analyses the residual stresses from the cladding was included in the integrity analyses. Accordingly the calculated stress intensity K_I was much higher than without these stresses. This request caused a remarkable reduction in the allowable T_k^a.

The operating utility Imatran Voima (present Fortum) decided to anneal the RPV in Loviisa, Unit 1 in 1996. The paper describes the procedures and requirements for qualifying the annealing in Loviisa. A comprehensive irradiation/annealing/re-irradiation qualification program by using a tailored weld as well as remnants from the surveillance program was carried out. The accelerated irradiation took place in a RIIAR research reactor (Corpus) in Dimitrovgrad. In addition a new surveillance program was planned and launched using the same above mentioned tailored weld and remnants from the surveillance program. Re-constitution welding was applied since only tested specimen halves were available from the original materials. Based on the test results from qualification program the annealing was finally qualified and carried out as planned in 1996 with success.

As a peculiarity a characteristic feature of the reactors in Loviisa NPP is that during some accident scenarios the RPV can be subjected to PTS caused by rapid cooling of the outside surface. This scenario could be launched in a possible Large Break Loca, when the ice-condenser would melt causing flooding of cold water in the reactor building. For such scenarios postulated crack behaviour had to be analysed.

- *Life management and operational life time extension at Paks NPP, T. Katona, A. Janosine, S. Ratkai, A Toth, SMIRT 17, 2003*

In Hungary there are 4 units of WWER 440/213 (2[nd] generation plants) in operation in Paks (Paks 1-4). The design life time of the units is 30 years. In order to extend the life time of the NPPs a licence renewal application has to be prepared not less than 5 years before the expiration of the design life time (EOL). In the year 2000 a study demonstrated the technical feasibility of PLEX as well as the economical rationale of the licence renewal. In the paper the licence renewal (LR) and activities for preparation of the PLEX are reported.

The ageing of SSCs relevant to safety was assessed already during periodic safety reviews (PSR) in the end of 1990's. Approach to prevent ageing was already implemented by modifications and replacements of some components and parts. Replacement of the turbine condenser tubes by stainless steel tubes is a good example. It allows increase of pH of the secondary circuit water chemistry, which in turn will reduce erosion corrosion of secondary components and piping and reduce deposition of magnetite on SG tubes.

A systematic lifetime management program has been launched at Paks. More than 23 components are included in the ageing management program (AMP). The main steps are at identification ageing effects, parameters of ageing process, ISI, testing and maintenance, acceptance criteria, condition monitoring, scheduling corrective actions replacements etc.

Since the RPV of Paks NPP were manufactured in Skoda Factories with due care for reducing of Cu and P content of the core welds, the neutron irradiation embrittlement is much slower and under better control than in the Russian 1[st] generation WWER 440 plants. It is assumed, that for the RPV of units 3 and 4 no extra measures will be needed for up to 50 years of safe operation. For Unit 2, the ECCS water needs to be heated up in order to ensure RPV integrity. For Unit 1 annealing of the core weld has to be considered with a 50% probability if operating up to 50 years.

- *Steps for lifetime evaluation and management of WWER main components, M. Brumovsky, J. Zdarek, NRI Rez, K. Pochman, NPP Dukovany R. Vitoch, Czech Power Company*

In Czech Republic bases for the evaluation of WWER components life time have been initialized. The 1st step has been preparation of Regulatory Requirements and procedures for life time assessment. A 2nd activity has been the elaboration of codes for reactor components by the Czech Association for Mechanical Engineers (ASI). Criteria for lifetime evaluation of RPVs, internals, procedure for RPV resistance against brittle fracture and for determination of neutron fluence have been prepared. Furthermore procedures for evaluation of defects found in ISI, PTS, mechanical testing of surveillance specimens, fatigue and corrosion damage evaluation, instrumented hardness and welding have been elaborated. The code will be divided in 5 sections:

1. Welding and brazing of components and piping
2. Characteristics of material
3. Strength calculations
4. Evaluation of residual life-time
5. Special material testing procedures

- *Lifetime management activities for NPPs in Czech Republic, M. Brumovsky, NRI Rez*

First some principle directions proposed by the ASI (Czech Association of Mechanical Engineers) relating to PLIM of Czech NPPs are described (as above in Paper 7). Then ageing management activities at Dukovany are discussed. The paper was published in 1997, so most of the activities have most probably already been completed. The RPV embrittlement problem and the PTS studies have been briefly described. The status regarding the embrittlement and integrity of the RPV internals in Dukovany NPP has also been described. Special attention has also been given to the life management of the steam generators, primary and secondary piping at Dukovany NPP. Finally the management of cable ageing has been dealt with. The paper is more or less "power point" version of the issue, and gives only general information and recommendations regarding the PLIM and does not give information on results obtained.

- *Program of plant life management at NPP Dukovany, K. Pochman, NPP Dukovany, M Brumovsky, M. Ruscak, NRI Rez*

Periodical lifetime evaluation on main components of NPPs is required by Czech Atomic law as well as by Safety Guides. The procedures for lifetime management have been given in the ASI Code, Section 5; "Lifetime determination of components and piping in WWER type NPPs". Lifetime management procedures for some selected components and piping (RPV, RPV Internals, Steam Generator (SG) tubes) at Dukovany has been described as examples of PLIM in the paper. The RPV embrittlement due to neutron irradiation is not of major issue at Dukovany, since the core weld has been made by clean filler metal with low impurity content such as P and Cu. The principles for PLIM of RPV internals are: detailed neutron calculations, evaluation of stresses and deformation of internals, irradiation and testing of material. As a result of the activities a lifetime will be proposed for the internals. The main ageing problems in SG are: SCC of collectors and heat exchange tubes and erosion corrosion of feed water pipes. For the SG tubes a procedure has been elaborated for evaluation of tube damage based on water chemistry variation and results from NDE of tubes. Based on these findings the life time of SG tubes can be predicted.

- *WWER Reactor Pressure Vessel design, M. Brumowvsky, NRI Rez, Yu. Dragunov, OKB Gidropress*

This paper is rather old and is describing the main design characteristics of the different types of WWER reactors. Special attention is given to the irradiation embrittlement problem of the 1st generation reactors WWER 440/230. The history of the brittle fracture analyses in Russia has been described briefly. The RPV integrity was first based on Fracture Analyses Diagram (FAD) which is based Charpy V test results and shifts of the transition temperature due to neutron irradiation, thermal embrittlement and fatigue. Later the 2nd generation reactors were designed according to new strength calculation norms based on Linear Elastic Fracture Mechanics approach (K_{Ic}) as in the ASME code.

The history of the higher than expected irradiation embrittlement rate of the core weld of the 1st generation reactors got special attention in the paper. According to the paper the irradiation embrittlement factor A_f could be up to 3.3 times higher than given in the design specifications and standards. The paper also describes the measures implemented for ensuring RPV integrity of the 1st generation plants. The main measures for ensuring integrity through the lifetime of the RPV have been the following:

1. correction of P/T curves for heat up/cool down and normal operation
2. heating the temperature of the ECCS water
3. changing of inlet nozzles of the ECCS system from cold leg to hot leg
4. installation of fast response valves between steam lines and main steam header
5. use of dummy elements and low leakage core
6. ISI
7. cutting samples from reactors for material testing and characterization
8. thermal annealing

11 Workshop Recommendations and Actions

- A revised subdivision with multiple allocation of publications *(participants)*
- Allocation of existing publications to new subdivision *(UvE, LD, MB)*
- 2nd round of consolidation in 2008 with all papers and the new subdivision *(UvE)*
- Unified keywords (terminology) for ODIN search and taxonomy *(MB, UvE)*
- Missing expert names should be sent to JRC *(participants)*
- A list of papers existing in ODIN should be distributed to the participants *(UvE)*
- All workshop material (summaries, viewgraphs, etc.) should be available in ODIN *(UvE)*
- Missing papers should be transmitted from the participants to the JRC for inclusion into ODIN *(participants)*
- A proposal for FP7 should be prepared *(UvE, LD, AB, VS, FG)*
- Joint effort in 2008 with the IAEA (2 workshops on WWER RPV embrittlement PLIM in Hungary and Ukraine); i.e. presentation of the knowledge consolidation within the 10 subdivisions of WWER RPV embrittlement
- The final goal is to cover the State-of-the-Art of WWER RPV embrittlement issues in a book (MB)

12 Annexes

Annex I

Agenda
1st Nuclear Knowledge Preservation & Consolidation (NKP&C) Workshop
WWER - WS1
Amsterdam, 11-13 December 2007

Date: 11/12/2007

- 12h00 Arrival and Lunch
- 14h00 Welcome and Planning (UvE)
- 14h15 COVERS – nuclear knowledge preservation activities (LD)
- 14h30 NULIFE – nuclear knowledge preservation activities (MB)
- 14h45 IAEA – nuclear knowledge preservation activities (KSK, MB)
- 15h00 Expression of Interest to DG RTD – Plans and Status (FG, UvE)
- 15h15 NKP&C at JRC (LD)
- 15h30 Coffee Break
- 16h00 General Info (UvE)
- 16h30 Subject: Chemistry – Presentation (AK)
- 17h00 Subject: Chemistry – Discussion (all, AK)
- 17h15 Closure Day 1 (MB, UvE)

Date: 12/12/2007

- 09h30 Subject: Annealing – Presentation (AC)
- 09h45 Subject: Annealing - Discussion (all, AC)
- 10h15 Subject: Modelling – Presentation (TW)
- 10h30 Subject: Modelling – Discussion (all, TW)
- 11h00 Coffee Break
- 11h30 Subject: Surveillance – Presentation (LK)
- 11h45 Subject: Surveillance – Discussion (all, LK)
- 12h15 Subject: Fracture Toughness – Presentation (AB)
- 12h30 Subject: Fracture Toughness - Discussion (all, AB)
- 13h00 Lunch Break
- 14h30 Subject: Testing and Microstructure – Presentation (FG)
- 14h45 Subject: Testing and Microstructure – Discussion (all, FG)
- 15h15 Subject: PLEX – Presentation (RA)
- 15h30 Subject: PLEX – Discussion (RA)
- 16h00 Coffee Break
- 16h30 Brainstorming (subdivision, experts, financing, future) (MB, UvE)
- 17h30 Closure Day 2 (MB, UvE)

Date: 13/12/2007

- 09h30 Subject: WWER Critical Issues and their Solving – Presentation (MB)
- 10h00 Subject: WWER Critical Issues and their Solving – Discussion (all, MB)
- 12h00 Conclusions and Closure Workshop (MB, LD, UvE)

Annex II

Experts

1st Nuclear Knowledge Preservation & Consolidation (NKP&C) Workshop
WWER - WS1
Amsterdam, 11-13 December 2007

Participants	*Interested*	
	V Petrosyan	Armenia
	T Kamenova	Bulgaria
J Brynda		Czech Republic
M Kytka		Czech Republic
M Brumovsky		Czech Republic
	J Kohopää	Finland
	M Valo	Finland
	W Daeuwel	Germany
	U Rindelhardt	Germany
H W Viehrig		Germany
F Gillemot		Hungary
	F Oszvald	Hungary
	A Horvaeth	Hungary
G Mariotti		Italy
	D Gilchrist	Italy
	G Aquilanti	Italy
A Chernobaeva		Russia
D Erak		Russia
A Kryukov		Russia
	V Nikolaev	Russia
	Tsniitmash	Russia
L Kupca		Slovak Republic
V Slugen		Slovak Republic
	A Hanzel	Slovak Republic
A Ballesteros		Spain
V Revka		Ukraine
T Williams		United Kingdom
K S Kang		IAEA
	J M Galan	OECD/NEA
	M Deffrennes	EC – DG RTD
	G van Goethem	EC – DG RTD
	W Hilden	EC – DG TREN
B Acosta		EC – DG JRC
R Ahlstrand		EC – DG JRC
L Debarberis		EC – DG JRC
U von Estorff		EC – DG JRC

European Commission

EUR 23718 EN – Joint Research Centre – Institute for Energy
Title: 1st Nuclear Knowledge Preservation and Consolidation (NKP&C) Workshop – WWER – WS1 – Summary Record
Author(s): U. von Estorff, M. Serrano, T. Williams, A. Ballesteros, M. Brumovsky, F. Gillemot, V. Slugen, K. Ilieva, A. Chernobaeva, R. Ahlstrand, L. Kupca

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Abstract
Nuclear knowledge had been build up continuously since the middle of the last century. After Chernobyl in 1986 the public opinion changed leading to a gradual phasing out process of nuclear energy in several Member States. During that time a trend at universities and in industry was observed of a decrease in students choosing nuclear related studies. Now the generation of senior nuclear experts is retiring. On the other hand, due to security supply and climate change issues (green house mitigation measures) receiving more importance lately, a renaissance of nuclear power is ongoing. In order to avoid a possible loss of capability and knowledge in the EU action is taken now preserving and disseminating it to the new generation.

There is a huge amount of information and knowledge available, either published or easily available, but also publications difficult to trace. Especially those are at risk of being dispersed or lost due to a series of factors, including:

- retirement of Senior Experts who were present at the time when most Nuclear Power Plants were designed and put into operation,
- generational gap (due to years of decline in new constructions, only a limited number of people started their career in that area)
- non-electronic publishing in the past
- limited dissemination possibilities
- language (many non-English publications from Eastern countries)

Therefore, the Institute for Energy of the Directorate General Joint Research Centre has developed a method for consolidation of nuclear knowledge.

The method relays on the mobilisation of all identified leading experts in the EU in re-evaluating old knowledge and consolidating what is necessary to create training materials for the new generations.

The methodology is applied for the present pilot study for consolidating and preserving WWER RPV safety related literature, which could be the start of a wider Nuclear Knowledge Preservation and Consolidation activity in the Nuclear Design Safety unit of the Institute for Energy.

Several reviewers received between 7 and 21 papers in their field of expertise, in order to review the content and present it for discussion and consolidation to the WWER Reactor Pressure Vessel embrittlement experts during the workshop.

The short-term scope is to reach a consolidated conclusion for the individual reviews, after the discussion and consolidation process during the workshop. The medium-term scope is a consolidated review in the individual expert fields. The long-term scope is to prepare a State-of-the-Art report for the complete WWER RPV Irradiation Embrittlement expert field, incl. the history and reasons of the choices made (material, composition, etc.). The last general document was produced more than 26 years ago by Nikolaev, Amaev and Alechenko, which is in Russian and needs upgrading.

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